Investigating and reassessment of the Bushehr NPP's SBO Analysis with presenting the last diesel generator recovery time

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Abstract:
Addressing the issue of station blackout accidents in the NPP is of utmost importance because they can compromise the protection layer of nuclear reactors and significantly impact core integrity. This study assesses the influence of operator actions and their consequences on the evaluation of accidents caused by station blackout (SBO) events in a Bushehr NPP using the Relap5 code. The goal is to determine the optimal recovery time for diesel generators. The operator's goal during an SBO accident is to manage and control the accident by depressurizing the primary circuit and using the accumulators to postpone the occurrence of core damage. Certainly, trying to activate emergency diesel generators or using transportable equipment, including transportable diesel generators, are some essential strategies for accident control and preventing core damage. But the important point at this stage is how long there is an opportunity to activate the diesel generators or use the transportable equipment to supply coolant for the reactor core and cool the second circuit to prevent core damage. In the case of a SBO accident at an NPP, two critical parameters must be considered for effective control. The first parameter is the selection of an appropriate strategy by the operator to take appropriate measures to reduce the pressure on the primary side of an NPP. The second parameter is knowledge of the time frame for the recovery of diesel generators and the subsequent activation of safety systems to manage and control the accident. According to the presented results, the actions of the operator significantly contribute to preventing SBO accidents. The activation of safety systems in this accident may not be enough to guarantee the safety of the NPP alone. However, the actions of the operator to continuously reduce the pressure of the first circuit, together with the operation of the safety systems, can lead to the establishment of safe conditions in the power plant.

Keywords: SBO- RELAP-BNPP-Recovery time

Abbreviation

ACC	Accumulator	LOOP	Loss Of Off-site Power
AC	alternating current	LPIS	Low-pressure injection system
BNPP	Bushehr Nuclear Power Plant	MCP	main coolant pumps
DC	Direct Current	NPP	Nuclear Power Plant
ECCS	emergency core cooling system	PSD	pulse safety valves
EDGs	emergency diesel generators	PWR	pressurized water reactors
EFW	emergency feed water system	RCP	reactor coolant pumps
EOPs	emergency operating procedures	RPV	reactor pressure vessel
FSAR	Final Safety Analysis Report	SBO	station blackout
HPIS	high-pressure injection system	SG	steam generators
LOCA	Lose Of Coolant Accident	SRD	steam relief device

1. Introduction

When a nuclear power plant (NPP) faces an accident, the control room staff must act quickly and follow the emergency operating procedures (EOPs) to restore the plant's safety and stability. They have been extensively trained on plant simulators to use the EOPs effectively. However, if the accident leads to core damage and possible radioactive release and the EOPs are no longer valid, it is crucial for the staff to depend on guidelines and technical support for effective handling of the situation. (Lee, Yilmaz et al. 2020).

Managing events is thus a vital element of the application of defense in depth. Managing accidents complements the operating procedures that shall be developed for normal conditions, anticipated operational occurrences, and accident conditions (IAEA 2008). The tsunami that hit the Fukushima nuclear plant triggered a series of events that led to damage to the operating reactors. The main cause of this damage was the loss of both onsite and offsite power during the long-term SBO, which disabled every operational system for cooling the core. Without a heat sink, the core of the reactor overheated, and the fuel structure melted down. The operators faced difficulties in using transportable equipment to supply feed water as coolant for the reactors due to the harsh environment. (Mehri, Safarzadeh et al. 2021).

A station blackout (SBO) is a situation where a nuclear power plant loses its connection to the external power grid, which can potentially trigger a severe accident. This was the case for the Fukushima NPP accident, which was caused by an SBO following a massive earthquake and tsunami. Therefore, it is crucial to study the causes and consequences of SBO events in nuclear power plants. (Li, Wang et al. 2014).

Operating in difficult situations due to a drastic external event or a serious accident necessitates the unified collaboration of all emergency staff, such as emergency response teams, security personnel and control room operators. (IAEA 2015).

The operator's role in accidents is crucial, as their awareness of the safety systems' operation affects their performance and behavior in managing and controlling the situation. Therefore, it is important to train and educate the operators on how to handle accidents effectively. Since the station blackout accident is very important to the safety analysis of NPPs in this research, a deterministic safety analysis of the station blackout accident is conducted in Bushehr NPP.

A station blackout accident is a loss of all AC power sources: both the normal operation power supply systems (such as the offsite grid and the turbine generator) and the emergency power supply like the diesel generators (Gjorgiev, Volkanovski et al. 2014). The SBO can be readily recognized by the loss of all operating systems, except the direct current (D.C) system (Volkanovski, Avila et al. 2016). The Bushehr NPP has hydro-accumulators in stages 1 and 2 as passive safety systems. Hydro-accumulators are tanks filled with water and pressurized gas that can inject water into the reactor core in the case of the LOCA. Hydro-accumulators could only postpone core damage following the SBO accident's occurrence (AEOI 2007).

When an SBO accident occurs, the operator can only rely on passive systems and pressurizer safety valves to control the NPP and postpone damage to the core. If the electricity power is connected by recovering auxiliary power sources such as diesel generators or using transportable diesel generators, the safety systems will be set up, and the operator will be able to control the accident and prevent core damage. But the important thing is: how long does the operator have the opportunity to prevent the accident from getting severe by connecting the auxiliary AC power of the power plant? The answer to this question indicates how much time is available to recover diesel generators or use transportable diesel generators and prevent the accident from getting severe. The primary objective of this analysis is to provide a guide for the operator so that, by knowing this time, he has the chance to reduce the consequences of the accident and subsequently prevent core damage.

This research simulates the station blackout accident scenarios of the BNPP using the Relap5/Mod3.2 code, a thermohydraulic tool, both with and in the absence of operator actions. The research determines the thermohydraulic parameters until the fuel-clad temperature reaches 1200 °C.

The operator's actions to reduce the primary pressure are analyzed as they affect accident management. These actions enable the use of the water stored in the passive safety systems of ECCS, which can delay damage to the core. The results of this research show that if at least two trains of safety systems are recovered before reaching the third critical point, damage to the core can be prevented during this accident.

2. Description of Bushehr NPP reactor

2.1- General information

The Bushehr NPP is a type of VVER reactor, which is a Russian version of PWRs. The reactor core has 163 hexagonal fuel assemblies that generate 3000 MWth of heat. The reactor has two circuits: the primary and the secondary. In the primary circuit, light water is utilized both as a coolant and moderator for facilitating heat transfer from the primary to the secondary circuit. The primary side also has a pressure vessel, a pressurizer, and four cooling loops with main coolant pumps (MCPs). The steam generators (SGs) are heat exchangers that connect the secondary and primary circuits. The secondary side produces steam from the heat of the primary circuit and sends it to the turbine. In Figure 1, the layout of the reactor and its components are shown.

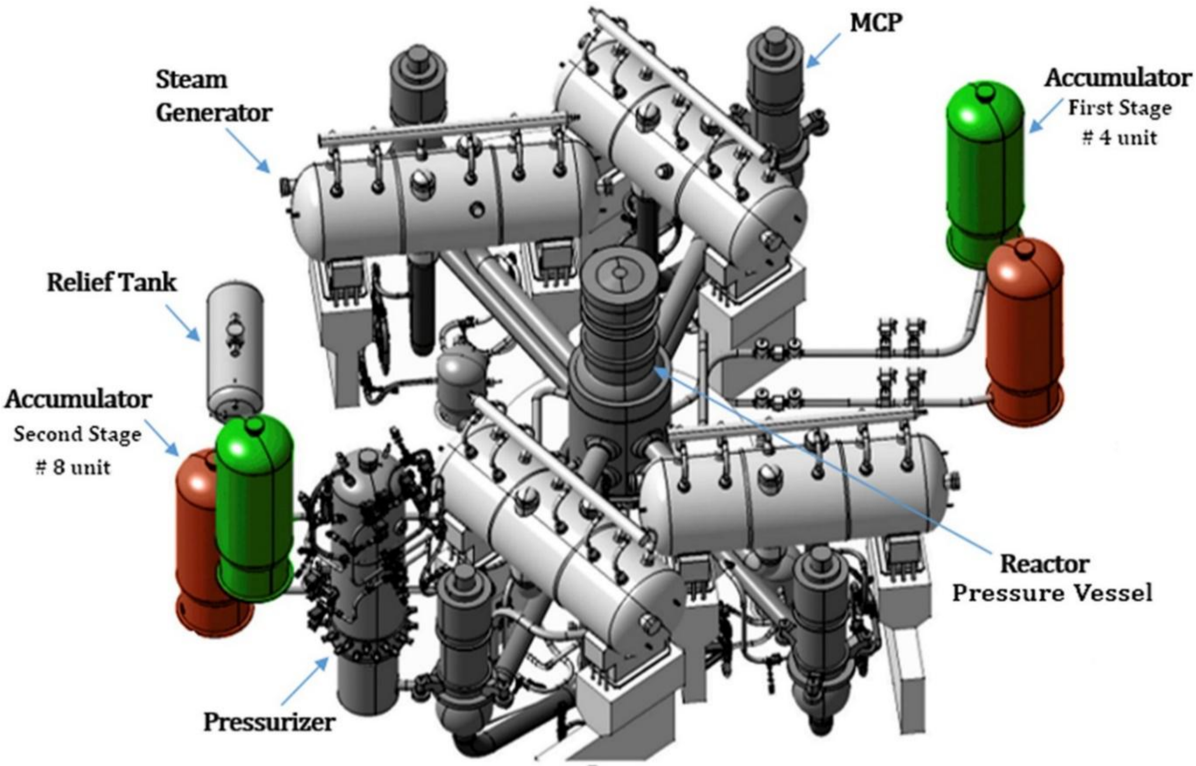


Fig 1. BNPP primary circuit (Hosseini, Shirani et al. 2020).

The primary circuit's ECCS includes a high-pressure injection system, a low-pressure injection system, hydro-accumulators (ACCs), and KWU tanks. These passive heat removal systems can inject coolant into the core in the loss of coolant accidents. The ECCS has two stages of accumulators in the BNPP. (Pouresgandar, Safarzadeh et al. 2022).

3. Description of the RELAP model

The RELAP5/MOD3 code is a thermal-hydraulic code that simulates light-water reactor systems. It was first released in 1979 and was used by INEL analysts to study LOFT and semi-scale experiments. Later, it became popular for analyzing commercial and experimental reactors and their scaled models. The RELAP5 (Fletcher and Schultz 1992) is the best-estimate code for simulating the thermal-hydraulic behavior of light water reactors. Of course, the main application of this code is the thermal-hydraulic analysis of the loss of coolant accidents in PWR reactors. It can analyze different types of accidents and transients that affect the reactor cooling system. It is mainly used for PWR reactors, but it can also handle research reactors, boiling water reactors (BWRs), and heavy water reactors. The RELAP5 is a trustworthy and versatile software for studying reactor safety and performance.

The RELAP5 code has many applications, such as validating regulations, verifying calculations, evaluating incident management methods, assessing user guidelines, and designing tests. The code can also simulate various thermal-hydraulic transients in any system (nuclear or non-nuclear) that contains steam, water, gas, and dissolved substances. In addition, this code can be used as a framework for analyzing nuclear power plants. The RELAP5 is a versatile software for analyzing the performance of the systems during transients and the reactor coolant system(Fletcher 1999).

The RELAP5 code employs a two-fluid model for simulating two-phase flow in nuclear power plants. Let’s delve into the specifics:

1. Break Flow Models:

- Ransom-Trapp Model: Initially used in RELAP5, this model had deficiencies and didn’t match experimental data well.
- Henry-Fauske Model: Introduced as an option alongside Ransom-Trapp, it performed better for various test cases. Eventually, it became the main modeling tool in RELAP5.
- Choking Model Shortcomings:
    - ✓ Two-Phase Critical Flow at Low Pressure: The default choking model could underestimate values significantly if the slip ratio isn’t forced to be nearly unity.
    - ✓ Subcooled Break Flow: For thin orifice plates near saturation point, the default model predicted critical flow values 40-50% less than experimental observations.
    - ✓ Thin Orifice Plate Configuration: Another shortcoming occurred at low pressure and low-quality conditions.

2. Two-Phase Flow Modeling:

   RELAP5 is based on a non-homogeneous and non-equilibrium hydrodynamic model for two-phase flow. It formulates equations temporarily averaged volumes and flow parameters (Fletcher and Schultz 1992).

A non-homogeneous two-phase model has been used to model BNPP in the RELAP. In addition, the abrupt area model has also been used, for example, in the break and the inlet and outlet nozzles of the reactor vessel. The choking model has been used in break modeling and safety valves. The cross-flow model has been used in the modeling of the steam generator, fracture, and inlet and outlet nozzles of the reactor vessel.

The BNPP model in RELAP5 is used to study the plant's behavior under different scenarios. The model is validated by comparing its results with the plant's data for steady-state and transient conditions (Mirzamohammadzadeh, Hadad et al. 2022).

The VVER-1000 RELAP5 model consists of the following components:

- The reactor core and vessel are divided into two sections with different power distribution coefficients (Kr).
- The pressurizer has three pulse safety valves (PSDs) that discharge into the same relief tank. One control valve opens at reactor pressures above 18.1 MPa, and two safety valves open at pressures above 18.6 MPa. These valves require DC power to operate.
- The safety components, which include the high-pressure injection system, eight tanks with boron supply, and the LPSI, also include a system model of the control volume.
- A control valve is located at the inlet of the secondary side, which consists of four steam generators with safety valves, four BRU-A and BRU-K valves, a steam pipeline, and a turbine.

In Figures 2 and 3 nodalization of the primary and secondary sides of the BNPP are presented. The average core channel is modeled using pipe component 30. The temperature of the hot rod in the hot channel is determined based on the maximum power peaking factor of the hot fuel assembly, which is modeled using pipe component 35.

The Table 1 contains details on the hydrodynamic considerations for the volumes of the reactor components.

Table1. The hydrodynamic considerations for the volumes of the reactor components.

Component	**Number**	**Hydrodynamic type**	**No. In nodalization**
The entrance to the reactor chamber	2	Branch	10-11
Downcomer	2	Annulus	15-16
vessel lower plenum	1	Branch	20
barrel lower plenum	1	Branch	21
Hot channel	1	Pipe	35
Average channel	1	Pipe	30
Bypass channel	3	Pipe	40-41-45
The upper part of the core	2	Branch	50-55
The lower part of the upper plenum	1	Branch	60
PTU	1	Branch	61

The upper part of PTU	1	Branch	70
Upper plenum	1	Branch	71
Vessel output	2	Branch	62-65

The pressurizer (910) is connected to the cold leg of the third loop through pipe 915 and valves 916 and 917. If pressure increases on the primary side, a signal is sent to these valves, causing water to be injected into the pressurizer. The pressurizer is also constantly connected to the hot leg of the second loop through pipe 905. This allows volume fluctuations in the primary side to be transferred to the pressurizer.

On the primary side of the horizontal steam generator, hot fluid enters through the hot collector. As it flows through the horizontal pipes, heat is exchanged with the lower temperature and pressure secondary side. This causes the primary fluid temperature to decrease before it exits the steam generator through the cold collector.

The VVER-1000/V-446 has a passive ECCS that includes Stage 1 and Stage 2 hydro accumulators. The Stage 1 accumulators (numbers 85, 87, 86, 88) are connected to the upper and lower plenums of the reactor pressure vessel. The Stage 2 accumulators (numbers 91, 92, 93, 94, 95, 96, 97, 98) are connected to the hot and cold legs of the main coolant pipelines.

The hot primary fluid enters the steam generator tubes through the hot collector. The tubes are modeled as five parallel channels, each with a different cross-sectional area but the same length. Each of these five channels is represented as an individual pipe in the code input. The cross-sectional area of each channel is proportional to the number of tubes in that particular channel. The main cooling line of the primary side has branches at the entrance (120) and exit (140). Downstream of those, there are five branches (141 to 145 and 121 to 125) that connect to the main coolant channels (pipes). Additionally, there are two upper branches (126 and 146) that connect to the emergency gas output system. The total water volume of the primary side is around 420 m^3. The secondary side of the steam generator is modeled using hydrodynamic components representing each flow channel. The downcomer section transfers the returning water from the steam-water separator (volume 508) to facilitate natural circulation within the steam generator. Adjacent hot and cold channels are connected through transverse connections.

The main feedwater supply to the steam generator is modeled using boundary volumes and connections 540 and 541. Separately, the emergency feedwater supply is represented using boundary volumes and connections 542 and 543.

The simulation also includes emergency cooling systems such as accumulator's stage 1 and KWU accumulators. These systems inject water into the reactor core and the hot and cold legs in case of an emergency. On the secondary side, the model considers the steam generator, vapor pipeline, generator power supply, emergency feed water system (EFW), and safety systems of the BRU-A and BRU-K based on the BNPP information. The VVER-1000 RELAP5 model is first validated in steady-state conditions. Table 2 shows conditions of actuation of the main safety protection system. The RELAP5 model is first validated in steady-state conditions.

Table 2. Conditions of actuation of the main safety protection system (AEOI 2007).

The triggering criteria for the primary protection systems and devices	Parameter value
HPIS activation condition	By a signal reaching difference less than 10 °C between saturation temperature in the reactor coolant system and temperature in hot leg.
LPIS activation condition	By a signal reaching difference less than 10 °C between saturation temperature in the reactor coolant system and temperature in hot leg.
Start-up of emergency feed water pumps	If certain signals coincide, such as a decrease in Steam Generator (SG) level by more than 900 mm or the coolant temperature in any hot leg of the primary loop exceeding 150°C.
Start-up of pumps TW (additional boron injection)	In the event of a reactor scram signal with a 4-second delay and a neutron flux level exceeding 10% of the value at the moment of the scram signal generation.
PRZ PSD Activation; Opening/closing pressure of control valve, MPa;	18.1/17.2
Opening/closing pressure of safety valves, MPa;	18.6/17.7
Time of PSD opening/closing, s	1.0/5.0
Actuation of SG PSD: Opening/closing pressure of control PSD, MPa;	8,24/6.87
Opening/closing pressure of control PSD, MPa	8.44/6.87
Actuation of MSIV:	$\Delta T_{S-1,2} \geq 75$ °C & PSG $\leq$ 4,9 MPa & $T_{1k} \geq 150$ °C

Table 3 compares the simulation results of the RELAP5 model with the FSAR reference data in steady-state conditions. The Relap5 code input model is validated by comparing the basic parameters of the nuclear reactor power plant under steady conditions with the design document data. (AEOI 2007).

Table 3. The basic parameters under steady state condition in BNPP

Parameter	FSAR (AEOI, 2007)	RELAP5	Error%
Core power (MW)	3000	3000	0
Pressure of the primary circuit (MPa)	15.7	15.67	0.2
Maximum temperature of Cladding outer surface (K)	625	631	0.1
Inlet coolant temperature (K)	564.8	564	0.1

Outlet coolant temperature (K)	598	596	0.33
Level of SG (m)	2.4	2.2	8.3
Pressure of steam in secondary circuit (MPa)	6.21	6.16	0.81
mass flow rate of steam at the outlet of the steam generator (kg/s)	408	412	0.81

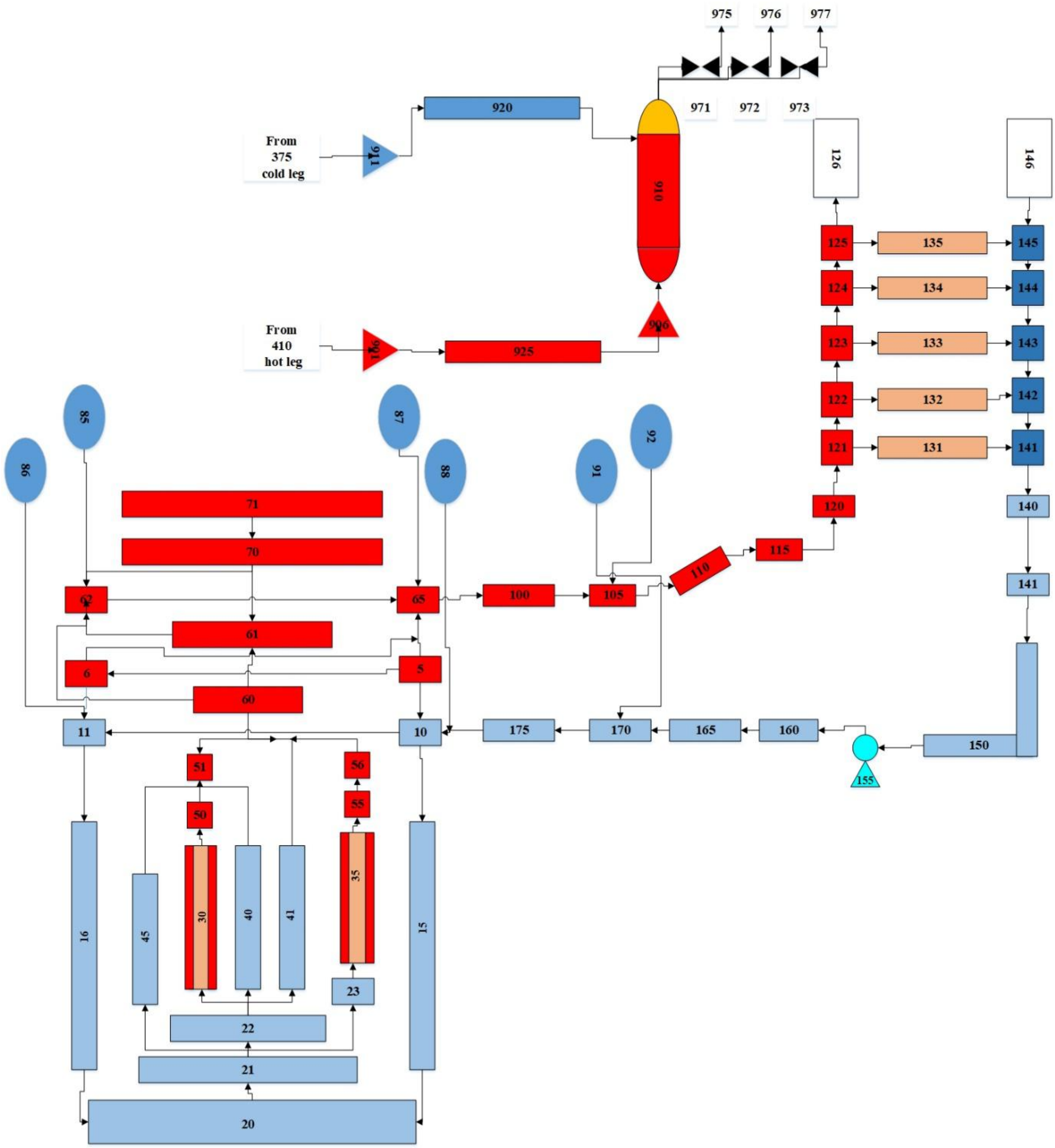


Fig 2. BNPP primary side nodalization.

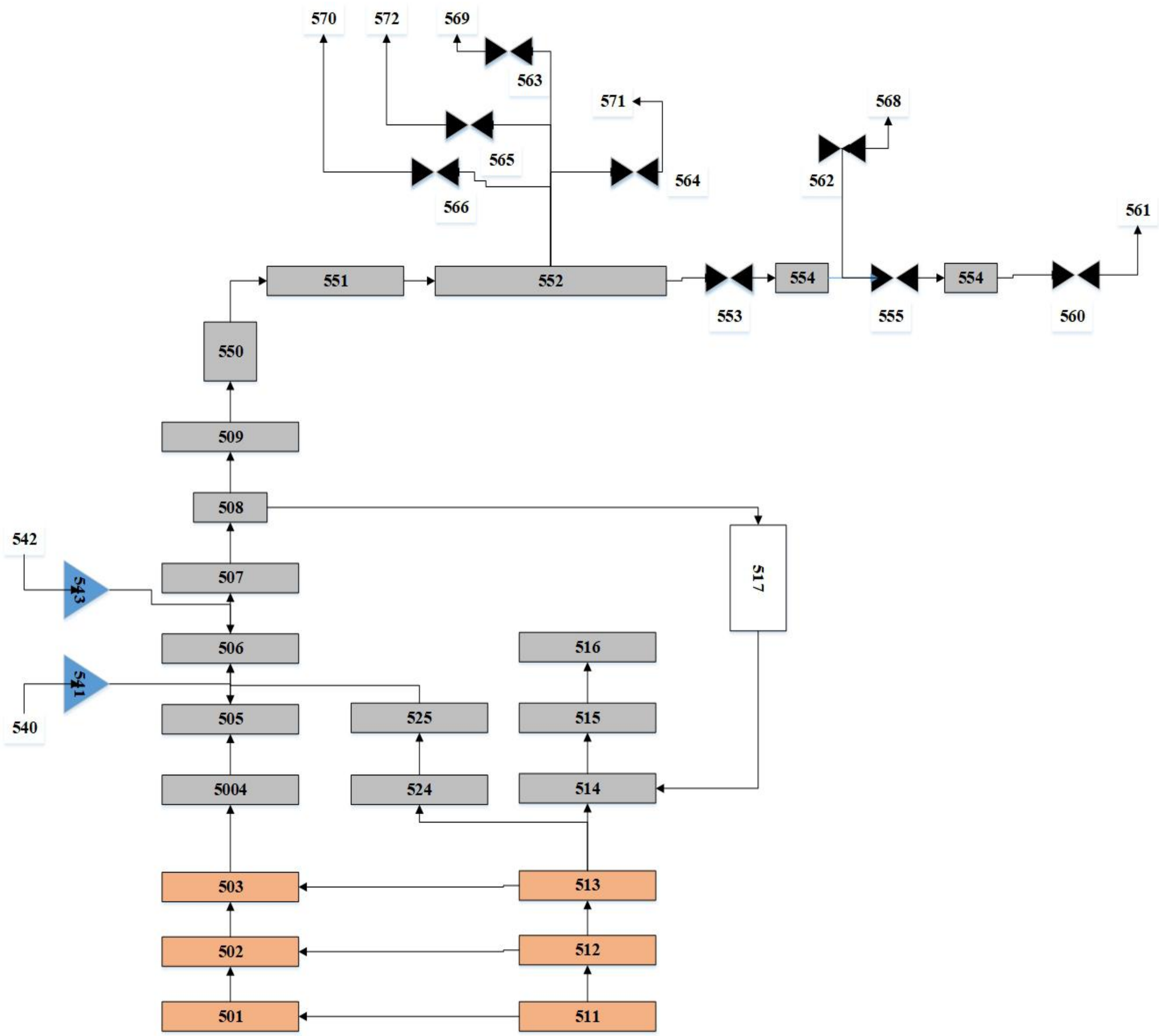


Fig 3. BNPP secondary side nodalization.

4. Definition of accident scenarios

When a station blackout happens, offsite power is lost (LOOP) and emergency diesel generators (EDGs) fail to start (Tatu and Kim 2017). This scenario is considered a design extension condition (DEC) (IAEA 2010).

The initiating events in this scenario involve the failure of both the diesel generators and the normal power supply sources, as well as the HPIS, LPIS (ECCSs), EFWS, and spray system. Different factors can lead to these failures, for example, natural disasters like earthquakes, tsunamis, military attacks, human errors, or inadequate maintenance (Amirsoltani, Pirouzmand et al. 2022). The only power sources left are DC power supplies from a class 1E

battery that need operator action. During SBO, in the reactor core, the decay heat is dissipated through natural convection within the primary side and transferred to the steam generators (SGs) on the secondary side. If the batteries are unavailable or depleted and AC power is not restored, the secondary cooling will fail, and the core will be damaged (Tatu and Kim 2017). This scenario is considered a design extension condition (DEC) (IAEA 2010).

The scenario also assumes that the primary coolant boundary remains intact. This is a conservative assumption that is not very realistic, but it is taken to study the worst-case scenario (AEOI 2007).

In the event of a station blackout, several things happen:

- The total AC power supply failure causes the RCPs, the main and auxiliary feed water equipment of the secondary side, and the primary side makeup and blowdown system to be tripped.
- The power source link for the pressurizer system power supply on the primary side and BRU-K on the secondary side is lost.
- The turbine stop valves are closed
- And scram signal generation.

During the initial stages of an accident, when there is evaporation of the steam generators (SGs) on the secondary side, BRU-A valves are automatically triggered to open and release excessive steam into the atmosphere. This controlled discharge helps to manage the pressure and prevent any potential damage. As the evaporation of the SGs continues, and they eventually dry out, the primary side experiences an increase in both coolant pressure and temperature. It is seen that steam generators drainage leads to heat-up of the core. (H. Ebrahimgol 2023).

Eventually, the pressure in the primary side reaches the setpoints where the pressurizer's safety valves. This is an emergency safety function to prevent the primary system from over-pressurizing. However, with the normal safety systems not working, this loss of primary coolant is problematic lead to the core and reactor vessel becoming damaged due to the continued heat buildup.

The sequence of SBO accidents has three crucial points. The first point is when the BRU-A valves are activated and steam generators lose steam. This reduces the heat removal from the primary side and damages the SG tubes. The coolant temperature on the primary side rises. The coolant level in the core falls, and the fuel rods begin to uncover at the second point. The fuel-clad temperature reaches 1200 °C at the third point.

The analysis of the reactor cooling system behavior and thermohydraulic parameters continues until the temperature of the clad reaches 1200 °C.

The core cooling is not steady, leading to core heat-up and fuel-clad temperatures exceeding 1200 °C. The strategy of pressure reduction and adding water from the accumulators has consequences, such as increasing hydrogen production. Specifications for safety system engineering are presented in Table 4.

Table4. Functional specifications of emergency protection system**(AEOI 2007).**

Conditions of actuation of the main protection systems and devices	Parameter value

Actuation of PRZ PSD; Number of control PRZ PSD, pcs Opening/closing pressure of control valve, MPa; Number of safety PRZ PSD, pcs Opening/closing pressure of safety valves, MPa; Time of PSD opening/closing, s	 1 18.1/17.2 2 18.6/17.7 1.0/5.0
Actuation of SG PSD: Number per one SG, pcs Opening/closing pressure of control PSD, MPa; Opening/closing pressure of control PSD, MPa Time of PSD opening/closing, s. Capacity with opening pressure, kg/s Characteristic ”flowrate-time of opening(closing)”	 2 8,24/6.87 8.44/6.87 1.0/1.0 258.0 Linear
Actuation of MSIV:	$\Delta T_{S-1,2} \geq 75$ °C & PSG ≤ 4,9 MPa & $T_{1k} \geq 150$ °C
BRU-A characteristics : Number per SG, pcs Control pressure, MPa Time of opening, s Capacity with opening pressure, t/h Characteristic ”flowrate-time of opening(closing)”	 1 6.67 15.0 900 Linear
Number of BRU-K valves, pcs The pressure of opening, MPa The pressure of closing, MPa Control pressure, MPa Capacity with opening pressure, t/h Characteristic ”flowrate-time of opening(closing)” Time of opening/closing, s	6 6.67 6.08 6.27 600 Linear 3.1

4.1- SBO accident analysis in the absence of operator actions (Scenario Version 1)

No personnel actions or additional safety system failures are assumed in this scenario. The primary and auxiliary feed water systems in the secondary circuit, the reactor primary circuit coolant pumps, the volume control and chemical systems, and the pressurizer are disconnected because of the complete loss of alternating current (AC) power sources. The BRU-K valve is isolated, and the vapor flow to the turbine is interrupted. The chronological progression of events during the Station Blackout accident in the absence of operator measures is shown in Table 5. In this scenario, the PSD valves function passively, meaning that the operator is not involved in any way. The valves open automatically when the pressure reaches the activation set point, and they do not require the electric power of the batteries to do so.

Table 5. The chronological progression of events during the Station Blackout accident in the absence of operator measures (AEOI 2007).

Time, s	Events	Interlocks, set point for actuation or other reason
0.0	Trip of all RCP sets Trip of the main and auxiliary feedwater systems of the secondary side Trip of makeup-blowdown system of the primary system BRU-K disconnection of PRZ	Loss of all power supply sources

	system power supply	
0.6	Shutting the stop valves of the turbine generator	Turbine emergency protection action
1.4	Generation of the Scram signal	NPP blackout
1.7	The onset of control rod motion	Emergency protection action
5.0	BRU-A opening	Reaching SG pressure of 7,15 MPa
2800.0	SG drainage	
7000.0	Onset of the core heat-up	
10000.0	End of calculation	

4.1.1- Simulation results for scenario version 1

The parameter variation throughout the accident is illustrated in Figures 4–9. The principal parameters are assessed with the FSAR (AEOI 2007) and previous scholarly work (Jabbari, Hadad et al. 2019). The power of the core decreases with the level of decay heat when the unit loses off-site power, and the reactor scrams are represented in Figure 4. The primary pressure slightly decreases as the inherent circulation of coolant on the primary side cools down the primary and secondary sides after the RCPs stop, as represented in Figure 5. This causes water evaporation and heat in the secondary circuit. The SG drains the water, as shown in Figure 6. Steam is discharged from the SG into the atmosphere through the BRU-A valve. The SG pressure is maintained between 7.15 and 6.27 MPa. Figure 7 shows the intermittent operation of the BRU-A valve, involving periodic closing and opening. As the evaporated feed water is not replaced by make-up water, the SG level and the BRU-A mass flow rate fall until the SG is empty of water.

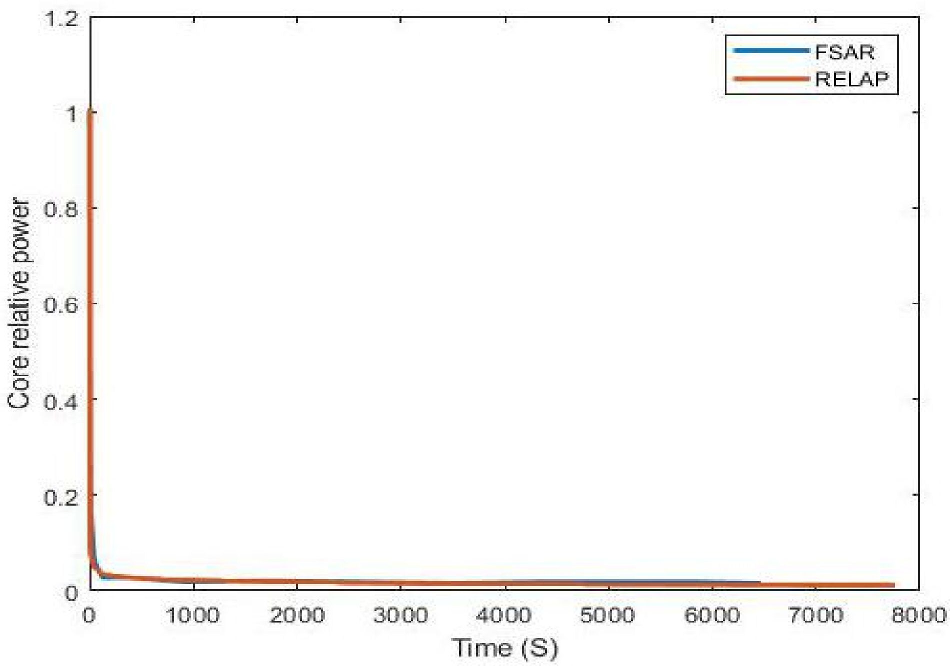


Fig 4. The core relative power in scenario version 1.

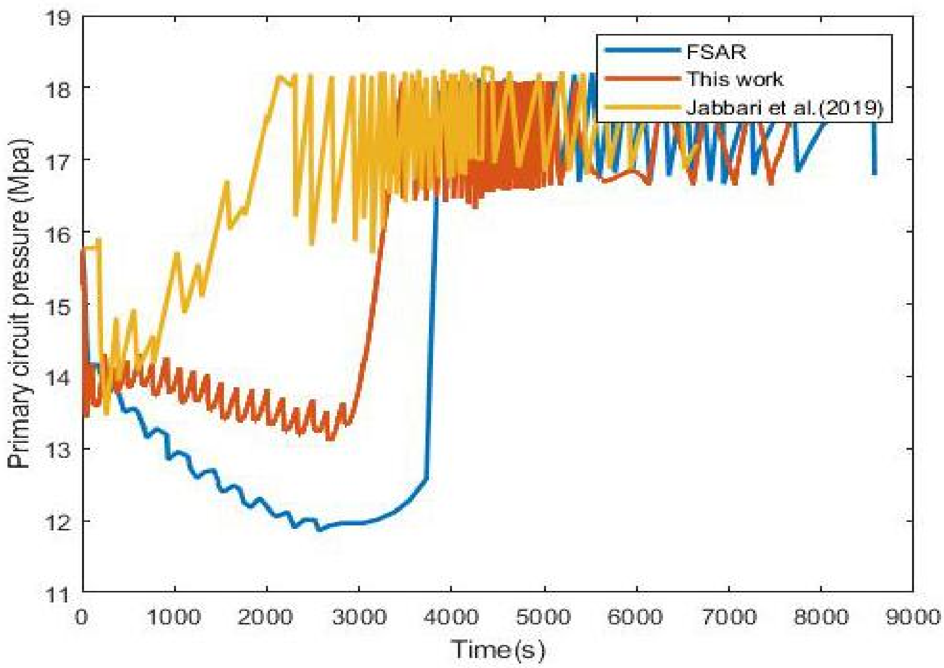


Fig 5. Core outlet pressure in scenario version 1.

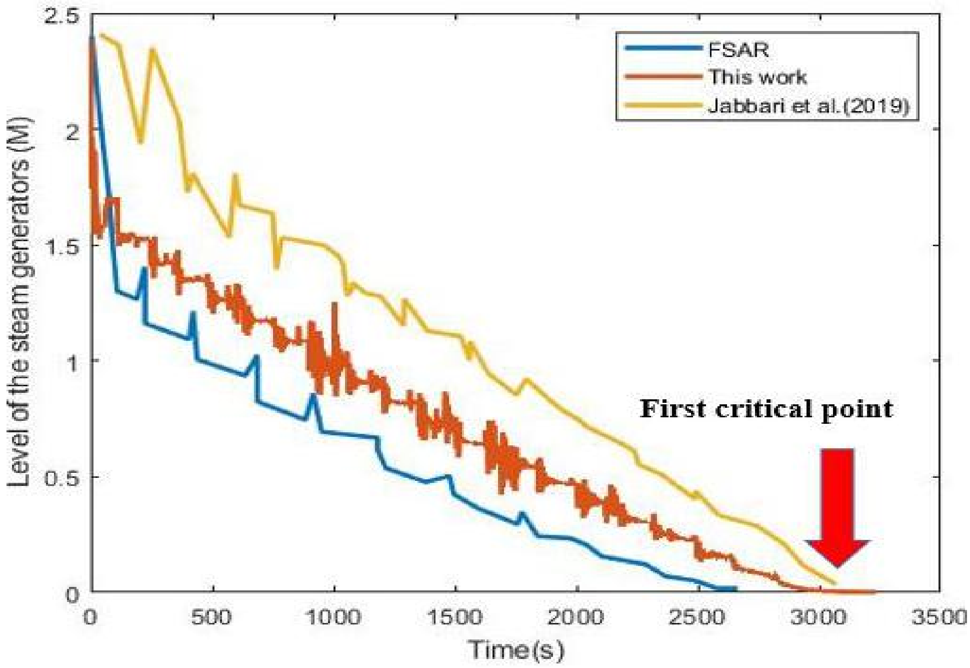


Fig 6. Water level inside the steam generators in scenario version 1.

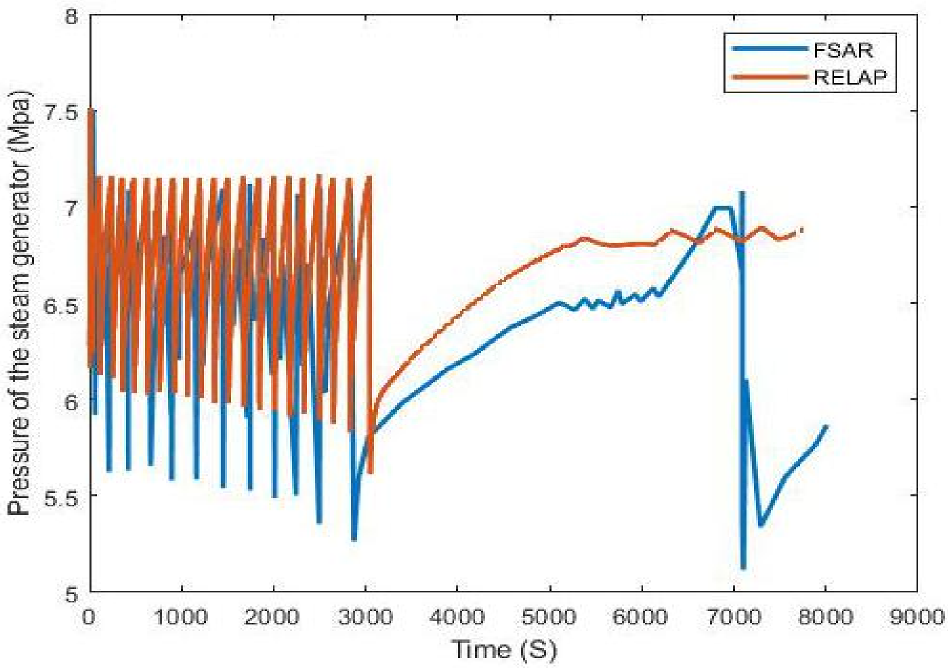


Fig 7. Secondary circuit pressure in scenario version 1.

As the initial critical point, the pressure on the primary side reaches the pressurizer PSD valve actuation level when the coolant in it gets heated up. The primary coolant is lost, and the core heats up as a result. As a result of reducing the coolant inventory in the core, the fuel rods dry out, which is a second critical point.
A third critical point occurs at 7300 s when the hot rod gets heated up, as illustrated in Figure 8. Figure 9 shows how the pressurizer level varied during the accident. The core and the reactor vessel will suffer damage at high pressure if no management measures are taken**.**

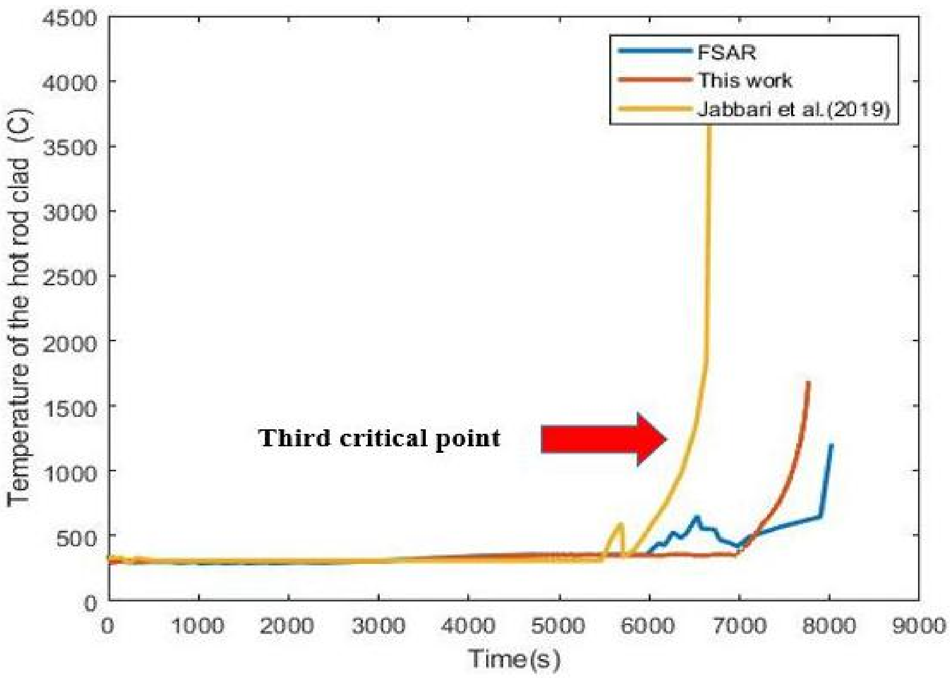


Fig 8. Temperature of the hot rod clad in scenario version 1.

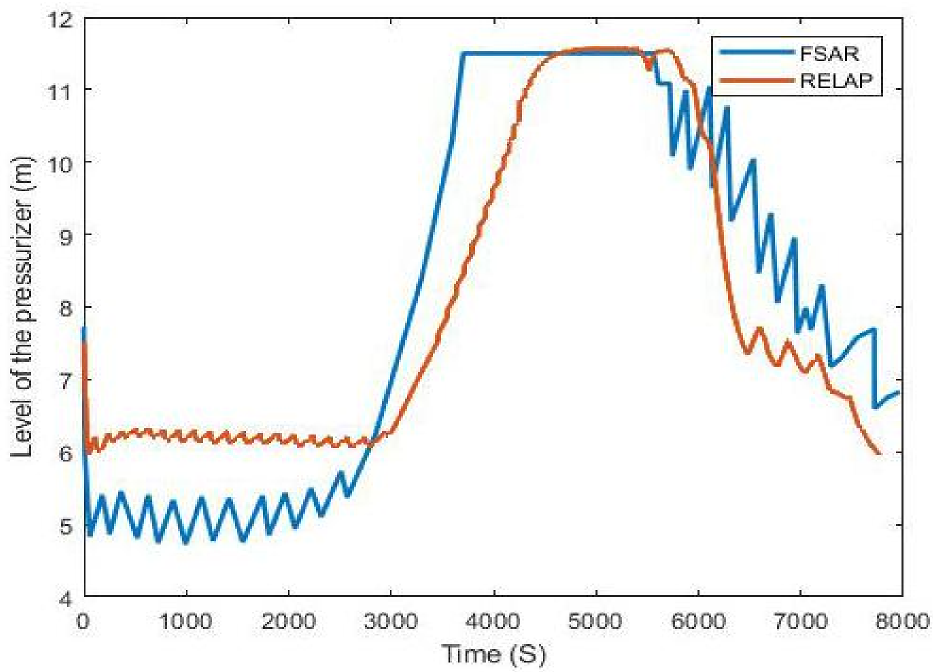


Fig 9. Level of the pressurizer in scenario version 1.

4.2- Station blackout accident analysis with consideration of operator actions (Scenario versions 2 and 3)

This scenario analyzes the operator's actions in accident handling to reduce primary pressure. The total AC power supply failure causes the reactor coolant pumps (RCPs), the primary and auxiliary feedwater equipment of the secondary circuit, and the makeup and blowdown system to trip. Table 6 shows the chronological sequence for scenario version 2, and Table 7 shows the chronological sequence for scenario version 3 of the events with system and device actuation.

The operator's role in this accident scenario is as follows:

- Reduce the pressure in the primary coolant circuit by opening the pressurizer PSD and gas removal valves.
- Continue decreasing the pressure until it reaches the level that triggers the activation of the hydro accumulators.
- Once the pressure has been sufficiently lowered, the hydro accumulators will automatically inject water into the primary side to help cool the reactor.

A decrease in the pressure of the primary side indicates that the operator has successfully opened the pressurizer PSD valves.

The operator’s actions are necessary for the reduction of the primary pressure and the possible use of water inventory in ACCs and Bushehr accumulators for core cooling during the SBO.

The core cooling is not steady, leading to core heat-up and fuel rod cladding temperatures exceeding 1200 °C.

Table 6. The chronological progression of events during the Station Blackout accident, taking into account the actions of the operator. (Scenario version 2) (AEOI 2007).

Time, s	Events	Interlocks, set point for actuation or other reason
0.0	Trip of all RCP sets Trip of the main and auxiliary feedwater systems of the secondary side Trip of makeup-blowdown system of the primary system BRU-K disconnection of PRZ system power supply	Loss of all power supply sources
0.6	Shutting the stop valves of the turbine generator	Turbine emergency protection action
1.4	Generation of the Scram signal	NPP blackout
1.7	The onset of control rod motion	Emergency protection action
5.0	BRU-A opening	Reaching SG pressure of 7,15 MPa
2800.0	SG drainage	
5000.0	Opening one PRZ PSD and the valves on the gas removal lines by an operator	
6150.0	Onset of the core heat-up	
8057.0	The onset of boric acid supply into the reactor vessel from the accumulators of stage one	Reaching the primary pressure of 5,88 MPa
8090.0	The outer surface of fuel rod cladding temperature exceeds the value of 1200 °C	
8100.0	End of calculation	

Table 7. The chronological progression of events during the Station Blackout accident, taking into account the actions of the operator. (Scenario version 3)(AEOI 2007).

Time,s FSAR	Time,s RELAP	Events	Interlocks, set point for actuation or other reason
0.0	0.0	Trip of all RCP sets Trip of the main and auxiliary feedwater systems of the secondary side Trip of makeup-blowdown system of the primary system BRU-K disconnection of PRZ system power supply	Loss of all power supply sources
0.6	0.6	Shutting the stop valves of the turbine generator	Turbine emergency protection action
1.4	1.4	Generation of the Scram signal	NPP blackout
1.7	1.7	The onset of control rod motion	Emergency protection action
5.0	5.1	BRU-A opening	Reaching SG pressure of 7,15 MPa
2800.0	3000	SG drainage	
5000.0	4200	Opening one PRZ PSD and the valves on the gas removal lines by an operator	
5550.0	5600	Onset of the core heat-up	

5842.0	5900	The onset of boric acid supply into the reactor vessel from the accumulators of stage one	dropping the primary circuit pressure of 5,88 MPa
6670.0	6500	The onset of boric acid supply into the reactor coolant system from KWU accumulators	Reaching the primary pressure of 2,5 MPa
7400.0	7550	The outer surface of fuel rod cladding temperature exceeds the value of 1200 °C	
7500.0	7800	End of calculation	

4.2.1- Simulation results for scenario version 2

Figures 10–15 show the analysis results of BNPP during a station blackout accident with the management of accident scenario version 2. The reactor scrambles after losing the unit's power sources. The power of the core decreases with the level of decay heat, as shown in Figure 10. The natural coolant circulation after the RCPs disconnect and stop slightly lowers the primary circuit pressure and evaporates the SG water inventory for 3000 s, as shown in Figure 11. The secondary circuit pressure cycles among the BRU-A opening and closing set points are shown in Figure 12. After the steam generator drains, the primary coolant heats up, and the pressure of the coolant rises to the pressurizer PSD valve's opening set point of 18.1 MPa shown in Figure 13.

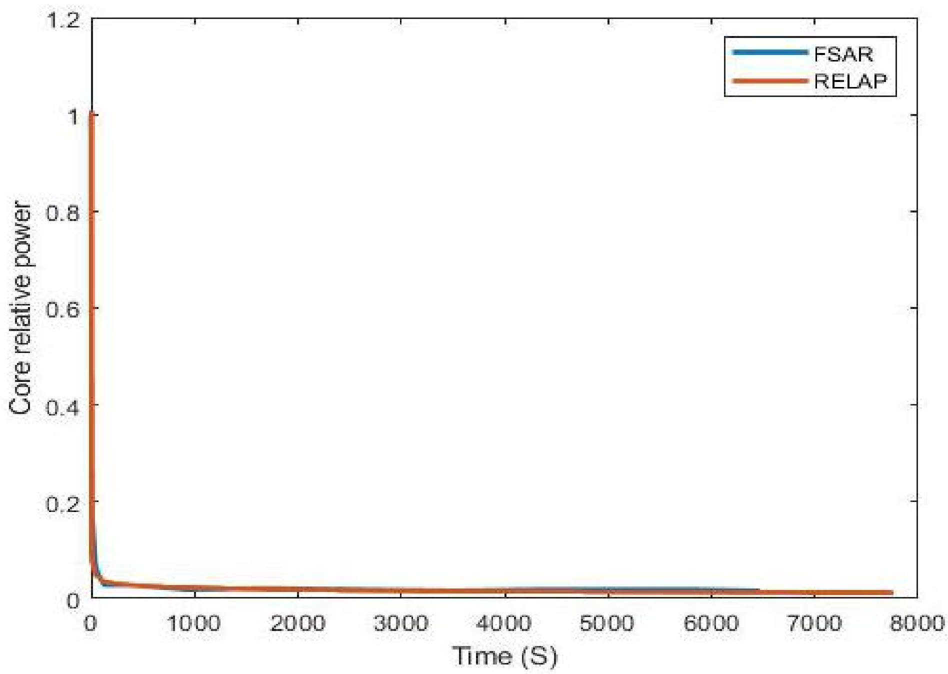


Fig 10. The core relative power in scenario version 2.

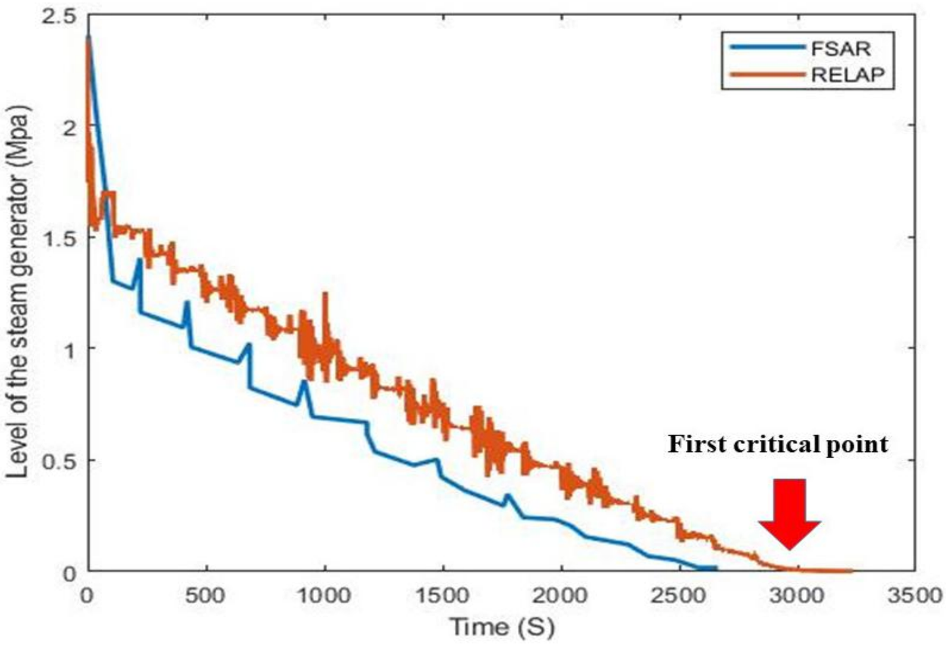


Fig 11. Water level inside the steam generators in scenario version 2.

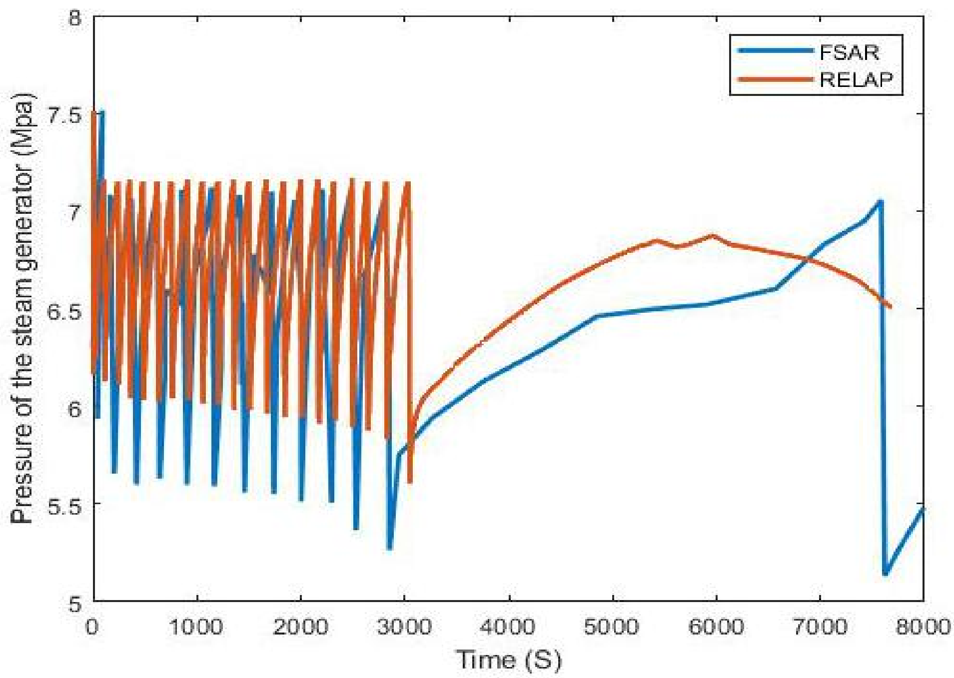


Fig 12. Secondary circuit pressure in scenario version 2.

The coolant loss within the core and the primary side arises from the PSD valve opening. During the blackout accident at 5000 s, the operator opens one PSD and the gas removal valve shown in Figure 13 to lower the primary side pressure and use the water resources in ACCs for the cooling process of the core. The primary side pressure drops to the ACCs actuation pressure. ACCs supply boron solution to the reactor vessel when pressure in the primary circuit reaches 5.88 MPa. The hot rod's temperature increase takes place at 7500 s, as shown in Figure 14. Figure 15 shows how the pressurizer level changes during the accident.

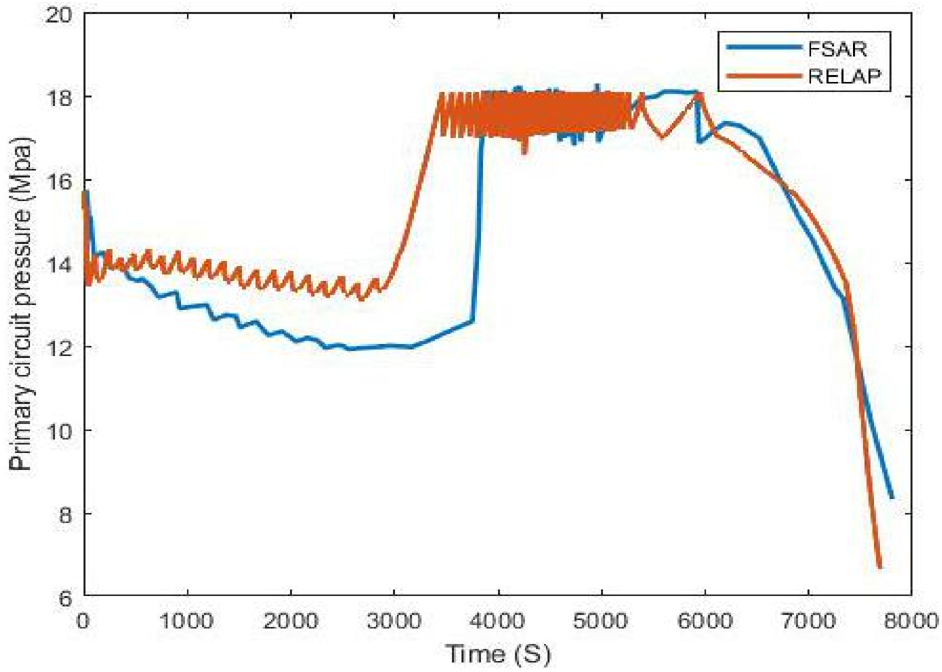


Fig 13. Core outlet pressure in scenario version 2.

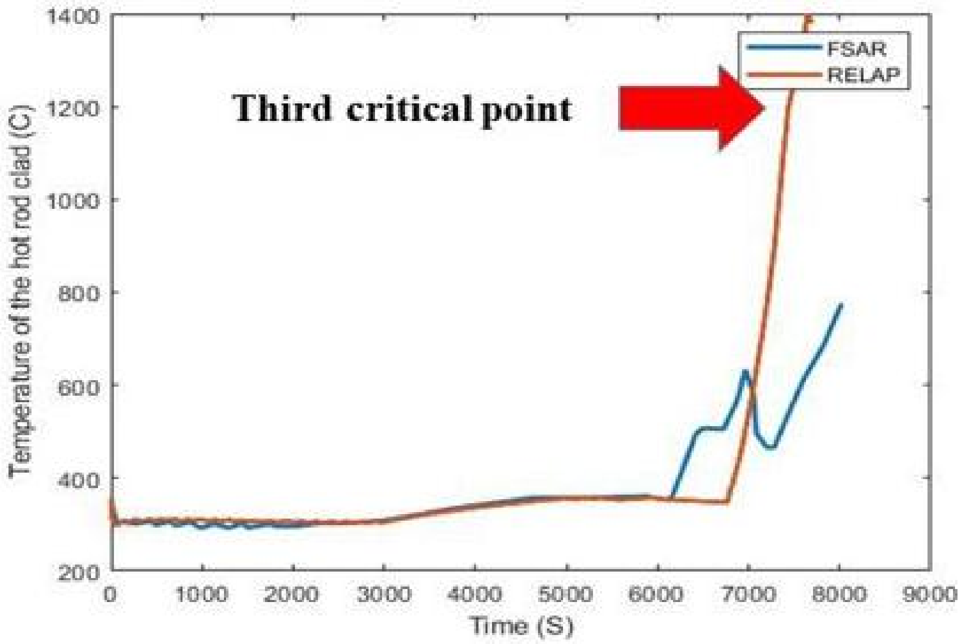


Fig 14. Temperature of the hot rod clad in scenario version 2.

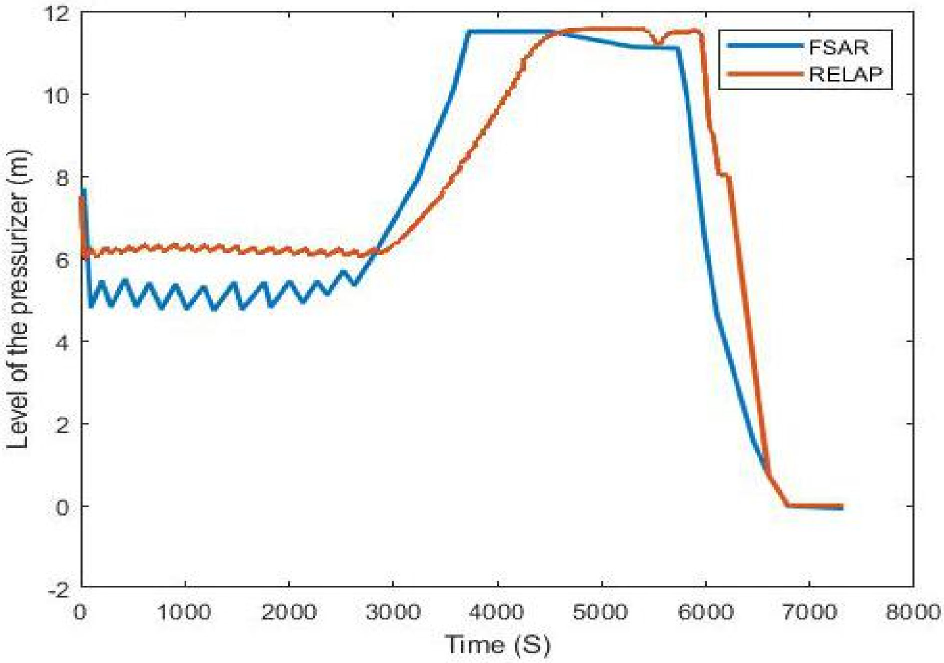


Fig 15. Level of the pressurizer in scenario version 2.

4.2.2- Simulation results for scenario version 3

Figures 16–21 show the analysis results of BNPP during a station blackout accident with the management of accident scenario version 3. The principal parameters are assessed with the FSAR (AEOI 2007) and previous scholarly work (Jabbari, Hadad et al. 2019). The power of the core decreases with the level of decay heat, as shown in Figure 16. Same as scenario version 2, the natural coolant circulation after the RCPs disconnect and stop slightly lowers the primary side pressure shown in Figure 17 and evaporates the inventory of SG water for 3000 s shown in Figure 18. The secondary side pressure cycles among the BRU-A opening and closing set points are shown in Figure 19. After the steam generator drains, the primary coolant heats, and the pressure of the coolant rises to the pressurizer PSD valve's opening set point of 18.1 MPa shown in Figure 17. The coolant loss in the vessel and on the primary side arises from the PSD valves opening. During the occurrence of the station blackout at 5000 s, the operator opens the three pressurizer PSD and gas removal valves shown in Figure 17 to lower the primary side pressure and use the water resources in ACCs and KWU accumulators for the cooling process of the core.

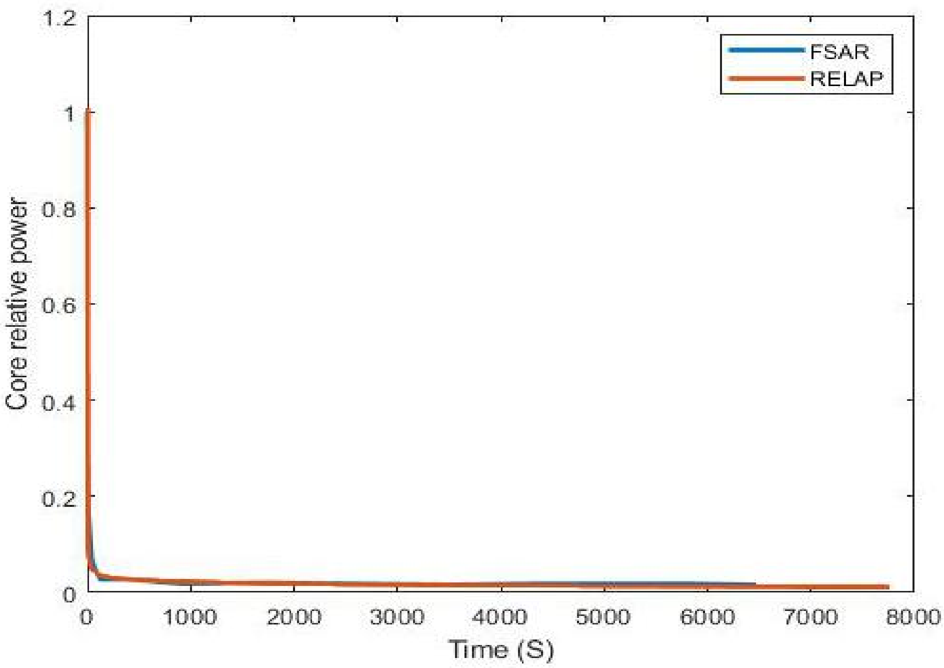


Fig 16. The core relative power in scenario version 3.

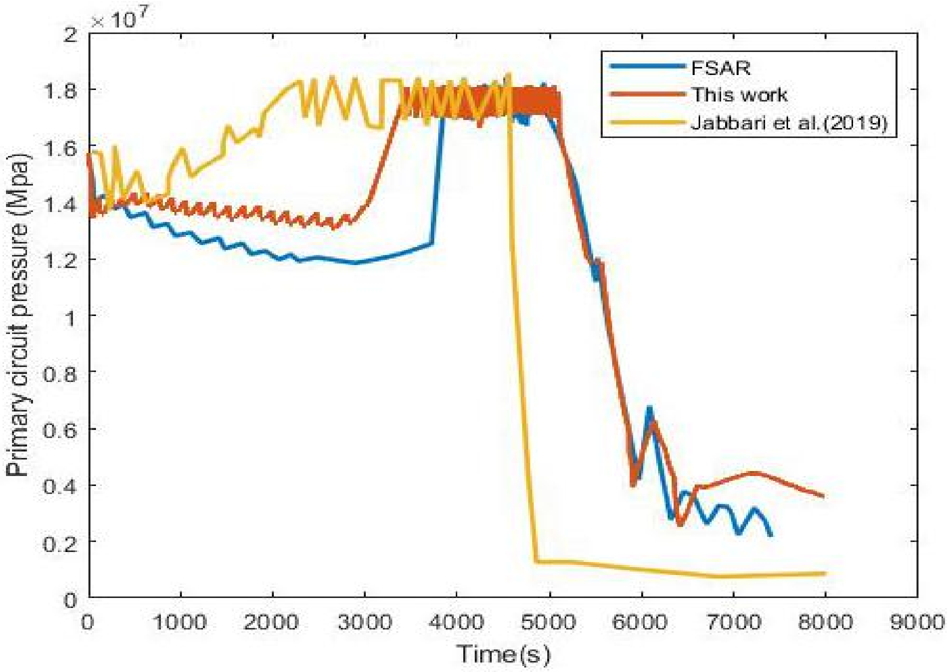


Fig 17. Core outlet pressure in scenario version 3.

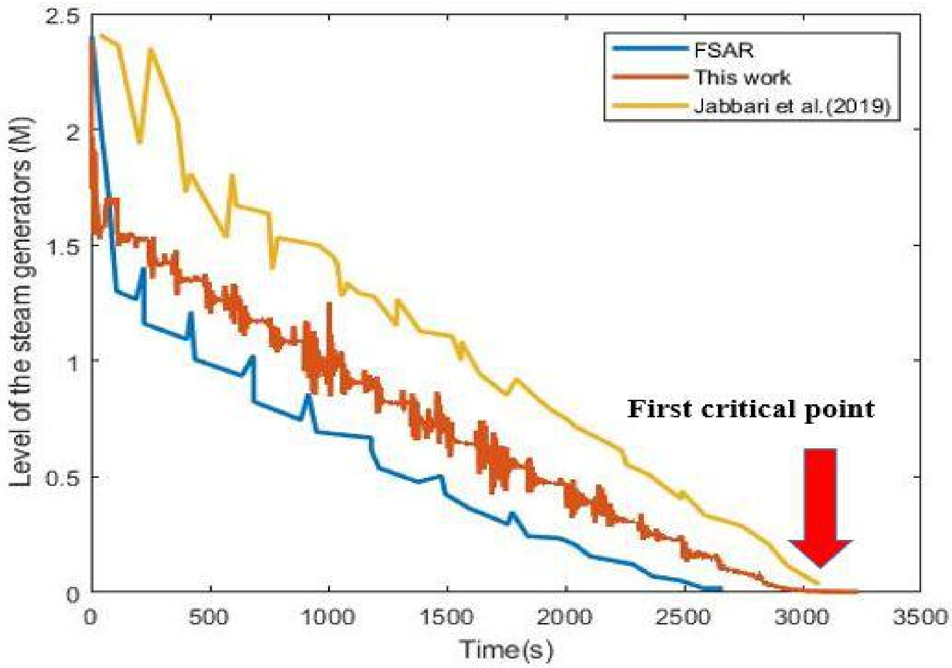


Fig 18. Water level inside the steam generators in scenario version 3.

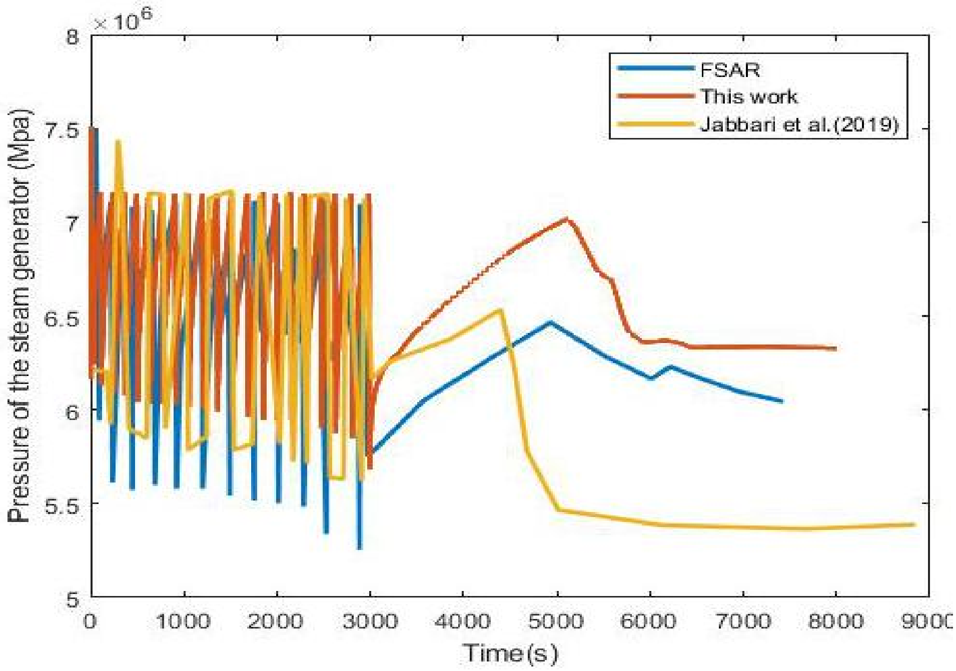


Fig 19. Secondary circuit pressure in scenario version 3.

The pressure on the primary side falls to the point of triggering the accumulators, and the boron solution is given from there. ACCs supply boron solution to the reactor vessel when the pressure reaches 5.88 MPa. The water level rises to above the fuel assemblies when the accumulators are empty. After some time has passed from the emptying of the accumulators and the subsequent reduction of the coolant level in the core, fuel rods dry out, and the fuel-clad temperatures increase, as shown in Figure 20.

Figure 21 shows how the pressurizer level changes during the accident.

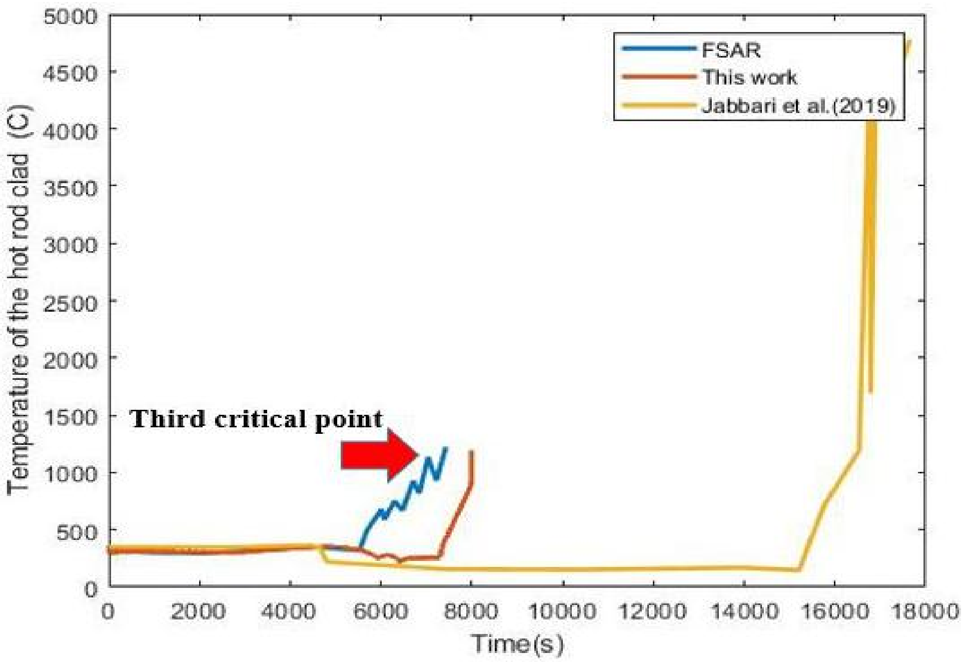


Fig 20. Temperature of the hot rod clad in scenario version 3.

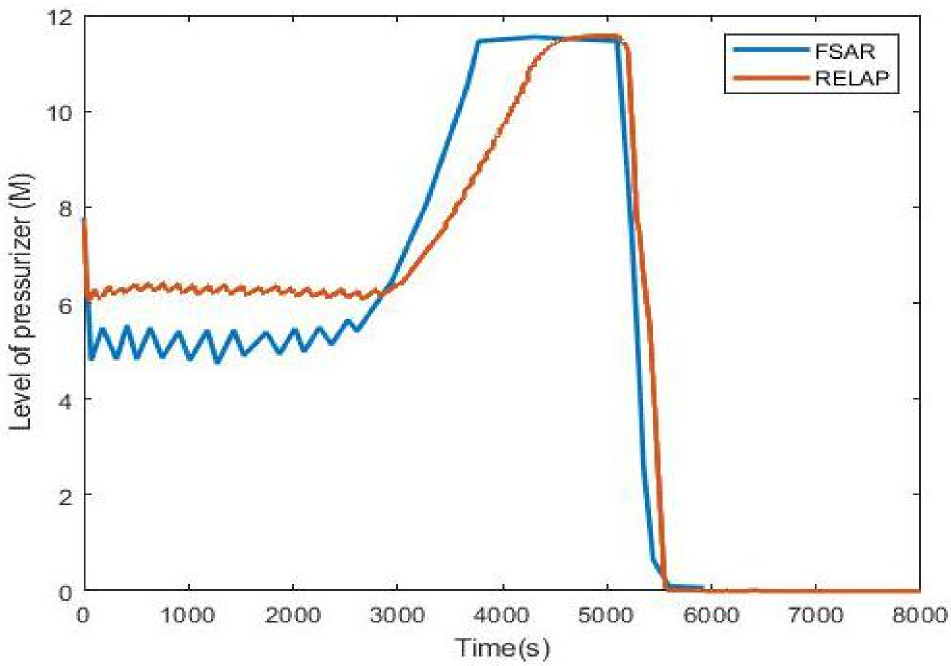


Fig 21. Level of the pressurizer in scenario version 3.

5. Ascertaining the latest time for diesel generator recovery

In the previously examined scenarios, the SBO accident without the return of the emergency a.c. power system and the prominent role of water accumulators are considered. However, in this section, the modeling of a simultaneous SBO and LOOP accident along with the return of emergency power by emergency diesel generators located in the NPP is discussed. To model the emergency power restoration, the systems that are restored after the power is connected are modeled by the RELAP5 code. For this purpose, in this section, the EFW systems and the HPIS and LPIS emergency injection systems are modeled by the RELAP code during this scenario.

Providing a guide for the plant operator and personnel is the goal of this section. The SBO accident resulted from a LOOP accident, with no activation of the emergency diesel generators. As shown in the previous sections, the operator's goal during an SBO accident is to manage and control the accident by depressurizing the primary circuit and using the accumulators to postpone the occurrence of core damage. The method of lowering the primary side pressure and adding water from the accumulators has consequences, such as increasing hydrogen production. Therefore, the operator encounters constraints in lowering the pressure of the primary circuit. Certainly, trying to activate emergency diesel generators or using transportable equipment, including transportable diesel generators, are some of the most important measures to control accidents and prevent core damage. These measures are carried out by the power plant personnel.

If the personnel are successful in activating the diesel generators or using the transportable diesel generators, two important and effective points should be considered: 1) the time when this equipment is put into operation and 2) the number of units that are successfully activated. But the important point at this stage is: how long is there an opportunity to activate the diesel generators or use the transportable equipment to inject coolant into the reactor core and cool the second circuit to prevent core damage? In the next step, the number of activated units will have a substantial effect on mitigating the accident.

In this section, the effective activation time of the equipment will be compared under two different assumptions about the number of activated pieces.

5.1- Determining the recovery time of diesel generators assuming all safety systems are activated (4 trains)

In the first step, we will examine the effective activation time in the first stage, where all the equipment of both circuits is successfully activated, for each of the three modeled scenarios. Figures 22–24 show the results of the successful activation of all the safety systems on the primary and secondary sides at different times in scenario versions 1–3. According to Figures 22–24, it is clear that the successful activation of all emergency diesel generators and the subsequent activation of all trains of emergency core cooling and emergency feed water systems on the reactor core for version 1 up to 7500 seconds, version 2 up to 7500 seconds, and version 3 up to 7800 seconds after the accident can prevent core damage, and after that, the accident cannot be controlled.

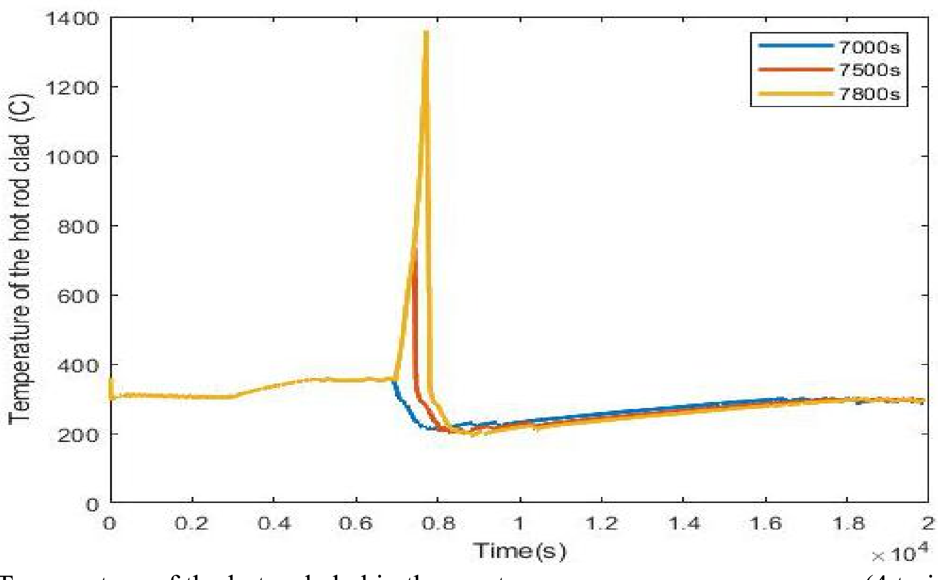


Fig 22. Temperature of the hot rod clad in the reactor emergency power recovery (4 train) at three different times after the accident in scenario version 1.

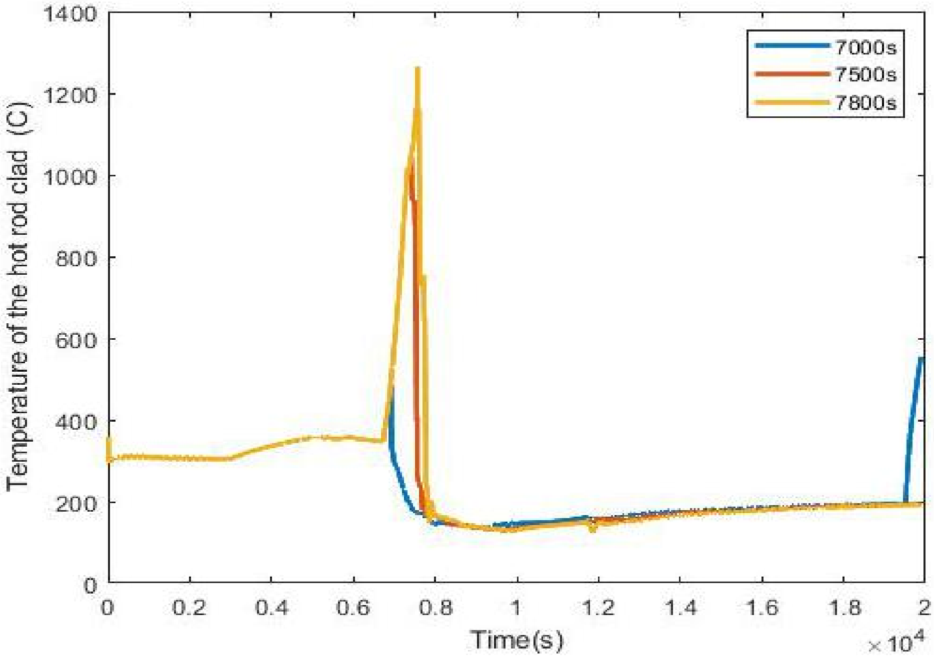


Fig 23. Temperature of the hot rod clad in the reactor emergency power recovery (4 train) at three different times after the accident in scenario version 2.

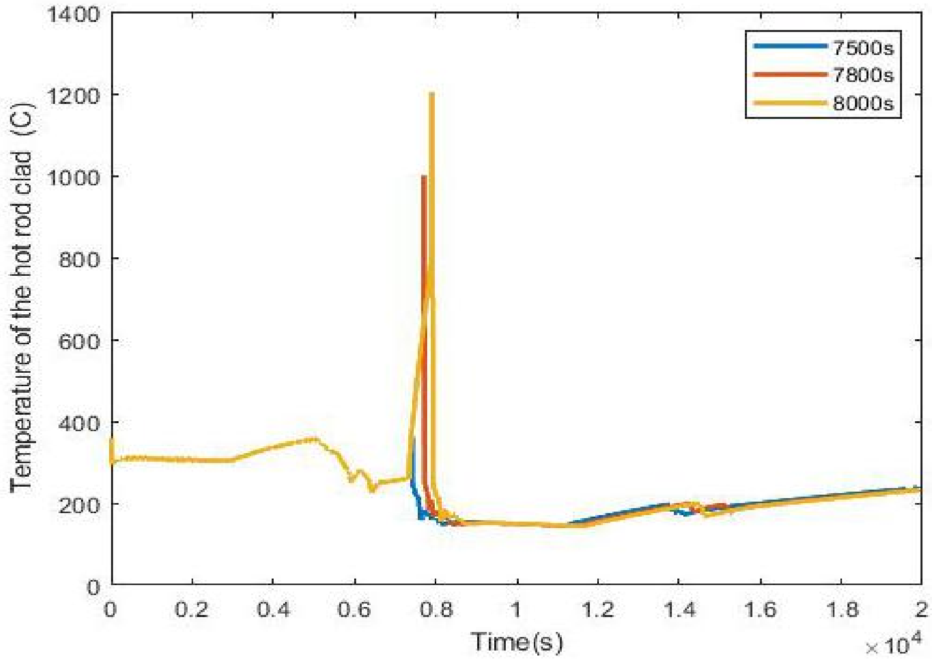


Fig 24. Temperature of the hot rod clad in the reactor emergency power recovery (4 train) at three different times after the accident in scenario version 3.

Figures 25–33 show the results of the successful activation of all the safety systems on the secondary and primary sides at 7500s for version 1 and version 2 and at 7800s for version 3 after the accident. The emergency feed water system in BNNP has the purpose of providing deionized water to the steam generators when there is a power loss that triggers the automatic shutdown of the reactor. The EFW system and the main BRU-A valve help to dissipate the remaining heat through the SG. This scenario assumes that all the safety system trains are recovered, and all four feed water trains are working. The SG water level will keep rising until it reaches 2.2 meters due to the fluid injection, as shown in Figures 25–27.

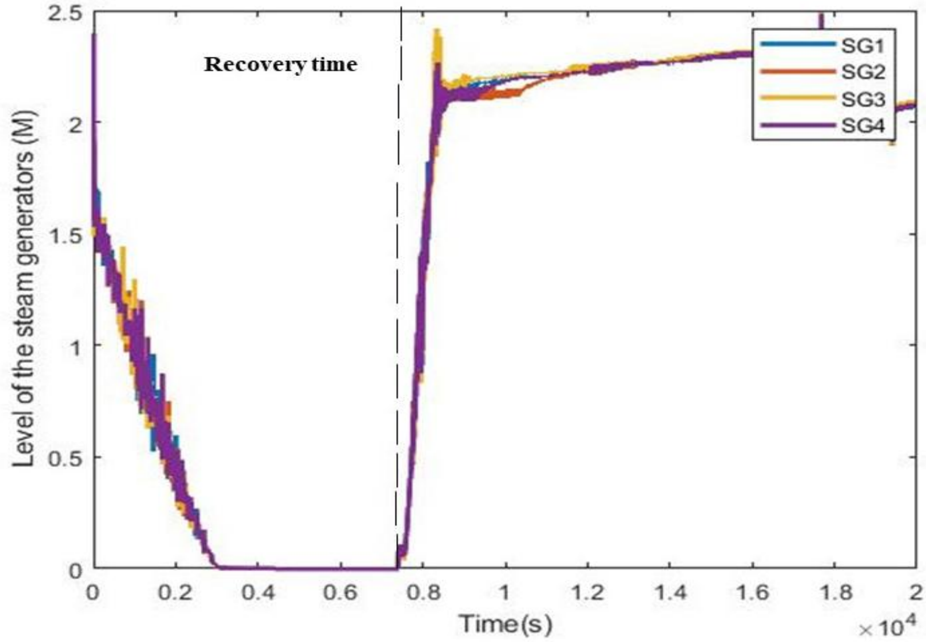


Fig 25. Water level changes in steam generators after emergency power restoration (4 trains) in scenario version 1.

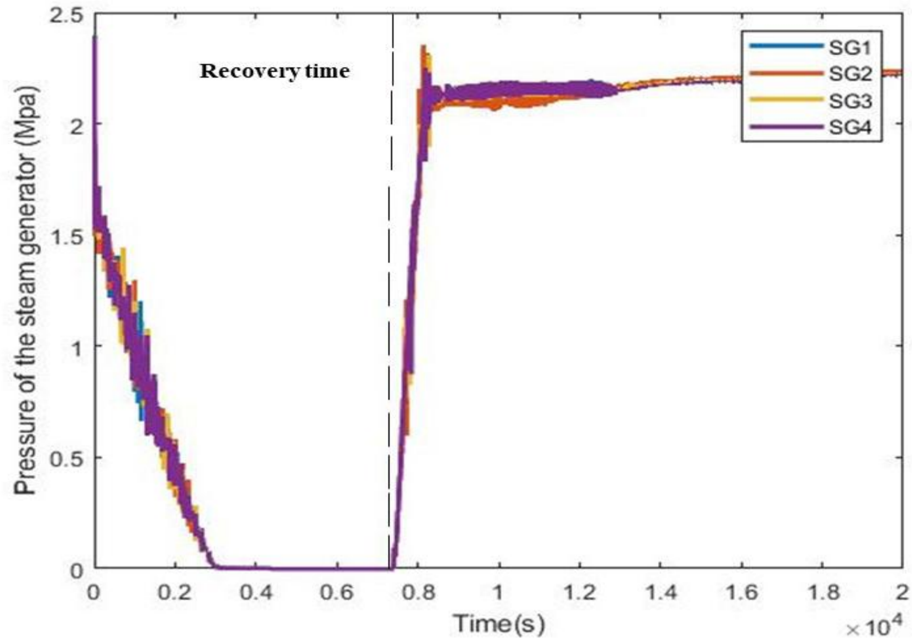


Fig 26. Water level changes in steam generators after emergency power restoration (4 trains) in scenario version 2.

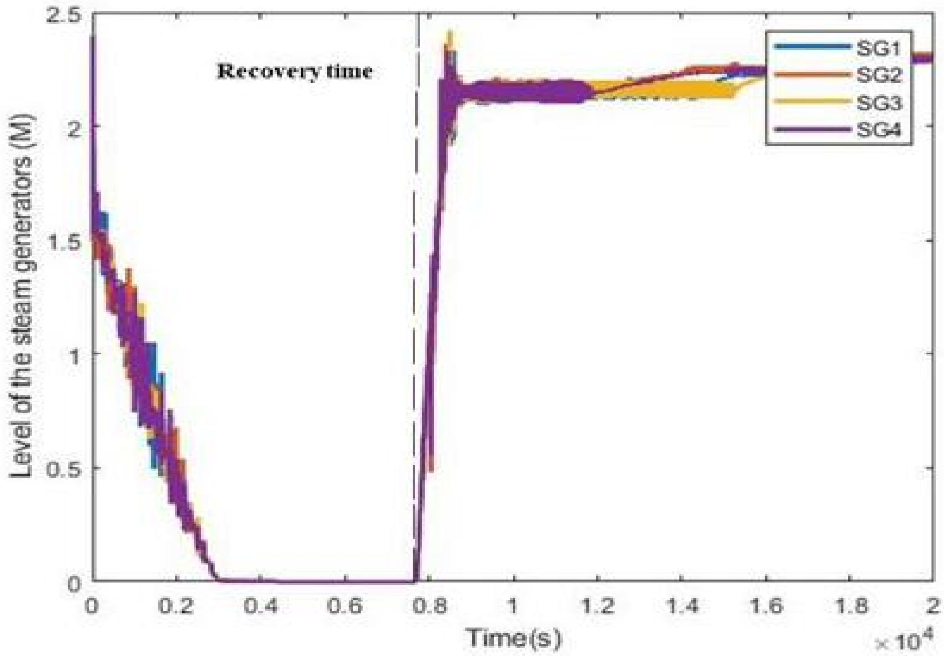


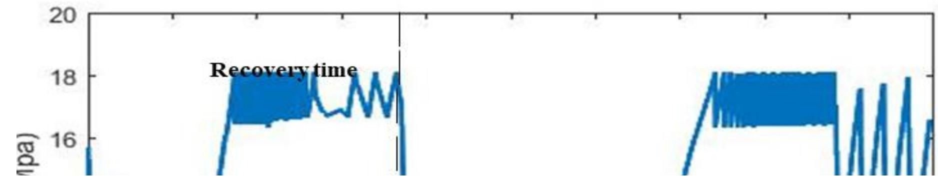
Fig 27. Water level changes in steam generators after emergency power restoration (4 trains) in scenario version 3.

As shown in Figures 28–30, the injection of Emergency Feedwater into the Steam Generator leads to a subsequent decrease in the primary pressure. Figures 28 shows that the activation of all four trains of safety systems led to an unexpected increase in pressure, which reached the activation set point of the pressurizer PSD valves, after a period of decrease. But what is the reason for this increase in pressure?

According to the process of various events, it can be concluded that before the safety systems were activated, the pressure in the first circuit was at the level of activation of the PSD pressurizer valves, and the heat transfer between the first and second circuits was not done due to the drying of the SGs. Therefore, as a result of the heat produced in the reactor core and the lack of heat exchange with the second circuit, the coolant in the core starts to boil and becomes biphasic. But with the implementation of safety systems such as HPIS in the first circuit and EFW in the second circuit, the pressure in the first circuit decreases as a result of the establishment of thermal transfer between the first and second circuits. This decrease in pressure, along with the injection of safety systems into the first circuit, reduces the temperature of the coolant in the core of the reactor. In simpler terms, lowering of the pressure in the first circuit reduces the saturation temperature of the cooling fluid in the core of the reactor. But after some time, the pressure on the first circuit starts to increase. This increase in pressure has several reasons, such as the injection of water into the first circuit through safety systems, which has led to an increase in the pressure in the first circuit. As it was said, the beginning of the thermal transfer between the first and second circuits has reduced the pressure, and at the same time, the injection of safety systems into the first circuit has increased the amount of coolant in the reactor core. As a result, by reducing the pressure caused by the thermal transfer between the first and second circuits, the saturation temperature of the coolant in the reactor core has also decreased. Considering that the reduction in pressure and subsequent reduction in the temperature of the reactor core coolant are not under the control of the operator and the operator has no involvement in this process, the reduction in the saturation temperature of the coolant due to the reduction in pressure

causes the process of boiling and two phases in the cooling liquid. Injection of safety systems, along with this factor, will increase the pressure in the first circuit.

As the level of water rises in the SG and the heat transfer between the first and second circuits increases, the second circuit pressure will increase to the limit of the BRU-A valve's performance, as shown in Figure 28.



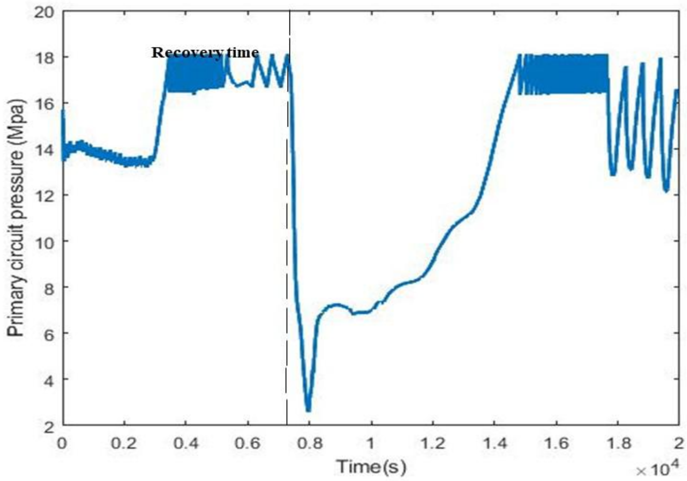


Fig 28. Primary circuit pressure after emergency power restoration (4 trains) in scenario version 1.

As shown in Figure 29, the injection of Emergency Feedwater into the Steam Generator leads to a subsequent decrease in the primary pressure. This version (version 2) considers that the operator lowers the primary side pressure with his actions, and the safety systems such as HPIS and fluid injection in the loop increase the pressure to about 4 MPa in the primary loop, and then the pressure drops gradually.

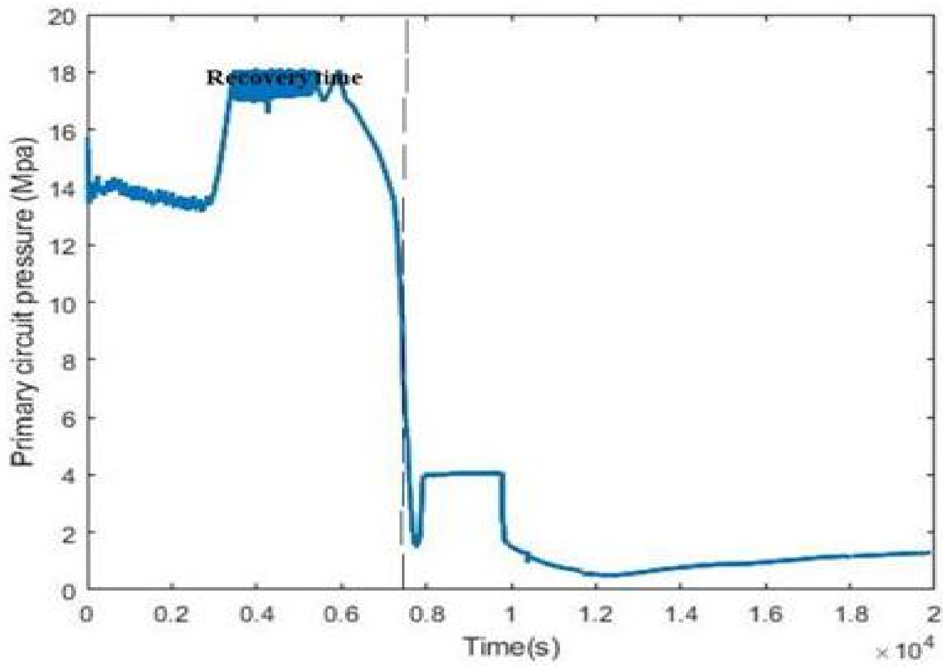


Fig 29. Primary circuit pressure after emergency power restoration (4 trains) in scenario version2.

The operator has reduced the pressure of the Primary circuit to the activation threshold of the first and second-stage accumulators with various actions in version 3 of the scenario. The pressure will rise slightly due to the safety systems and the fluid injection into the Primary circuit, and then it will fall slowly again due to the heat transfer between the primary and secondary circuits, shown in Figure 30.

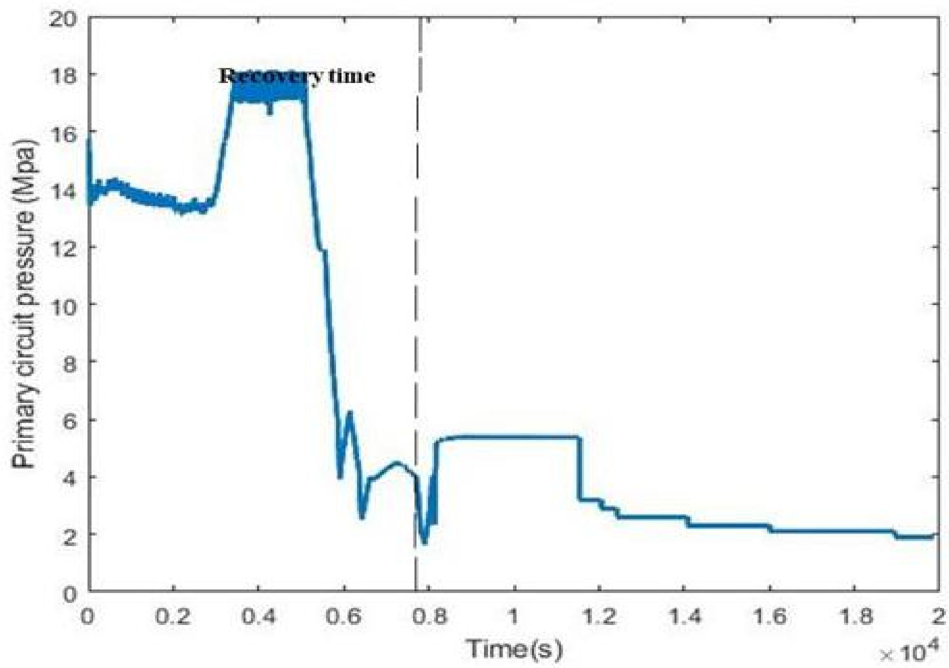


Fig 30. Primary circuit pressure after emergency power restoration (4 trains) in scenario version3.

As the water level rises in the steam generators and the heat transfer between the first and second circuits increases, in the scenario version 1, due to the increase in the pressure of the primary circuit (figure 28), the pressure of the second circuit has also increased, as shown in figure 31.

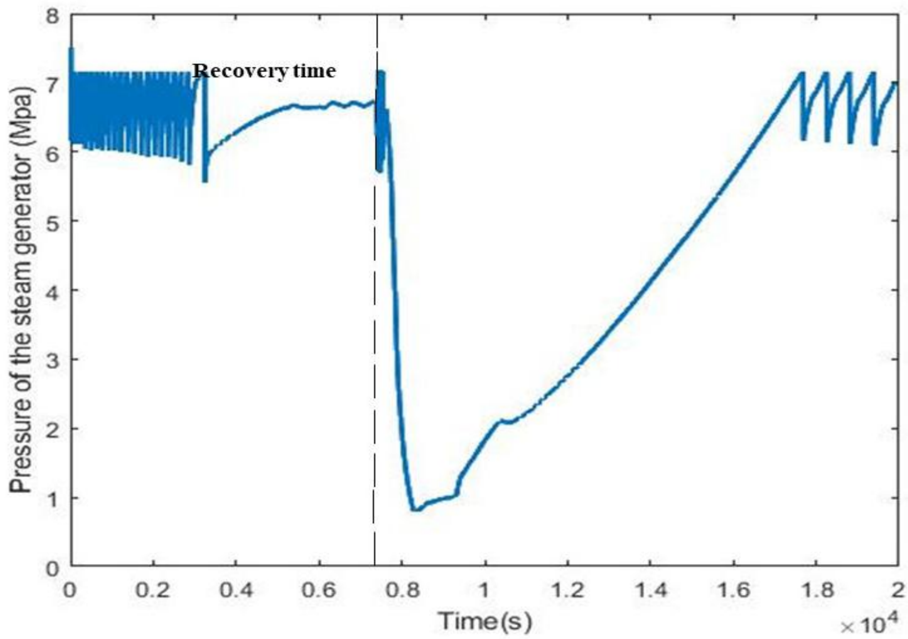


Fig 31. Steam generators pressure after emergency power restoration (4 trains) in scenario version 1.

As the water level rises in the steam generators and the heat transfer between the first and second circuits increases, the pressure in the primary and secondary circuits will both decrease steadily due to the operator's actions (scenarios versions 2 and 3) to lower the pressure in the primary circuit, and the continuous downward trend in the pressure in the primary circuit is shown in Figures 32-33.

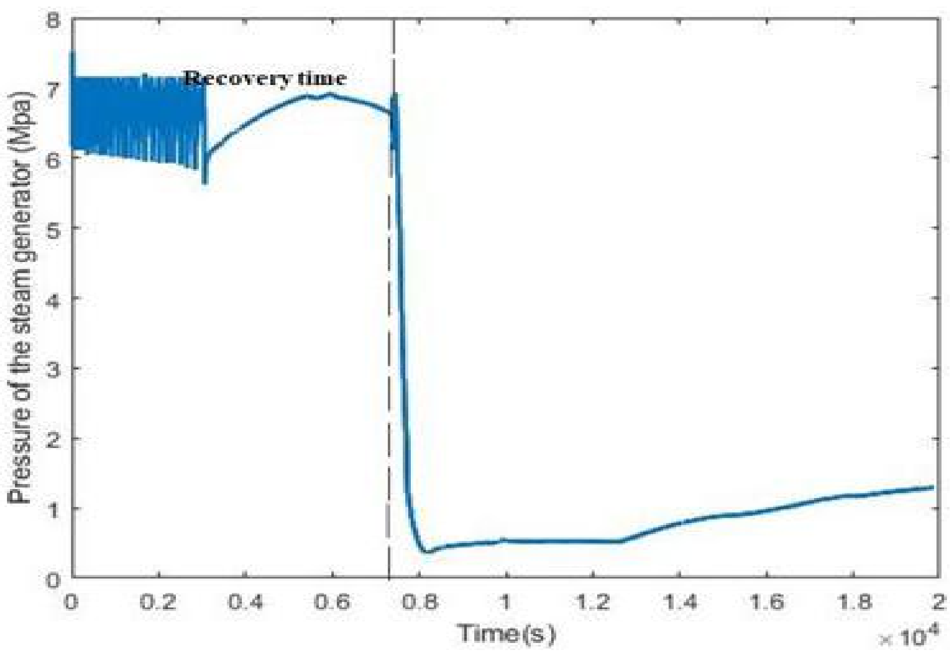


Fig 32. Steam generators pressure after emergency power restoration (4 trains) in scenario version 2.

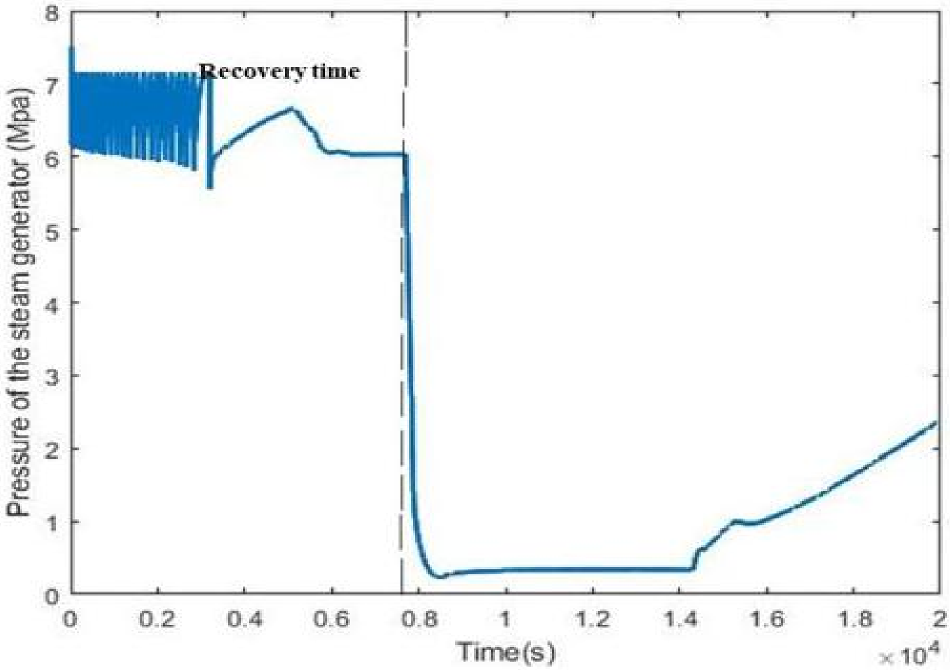


Fig 33. Steam generators pressure after emergency power restoration (4 trains) in scenario version 3.

5.2- Determining the recovery time of diesel generators with the two train safety systems activated

In the second step, we will evaluate the effective activation time of the equipment, where only two trains of equipment are successfully activated, with a conservative assumption.
Figures 34–36 show the results of the successful activation of two trains and the safety equipment of the primary and secondary circuits at different times in scenarios 1-3. According to the results, it is clear that the successful activation of two trains of emergency diesel generators and the subsequent activation of two trains of emergency core cooling and emergency feed water systems on the reactor core for up to 7500 seconds for versions 1-2 and up to 7800 seconds for version 3 after the accident can prevent core damage, and after that accident, it cannot be controlled.

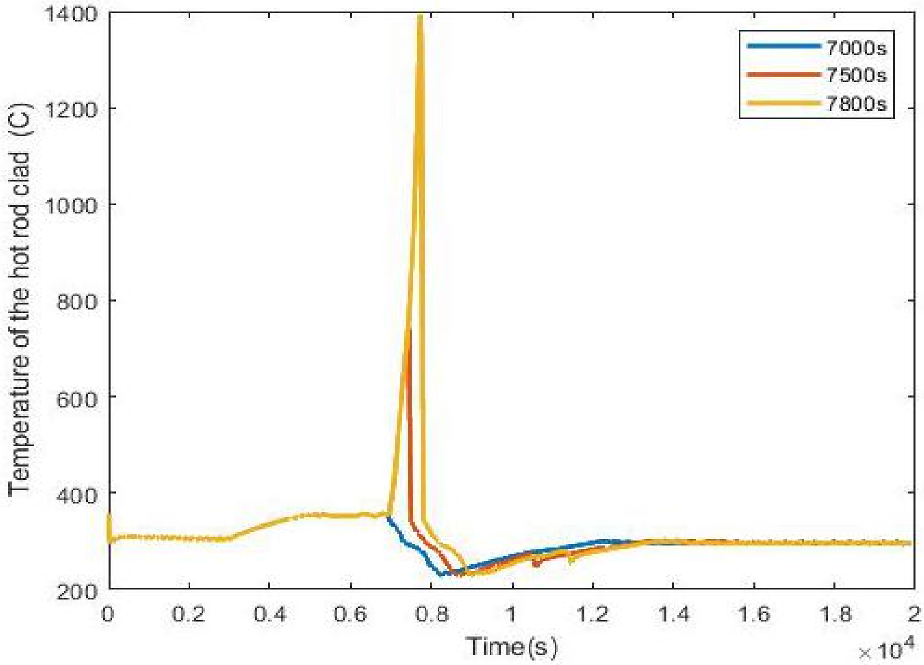


Fig 34. Temperature of the hot rod clad in the reactor emergency power recovery (2 train) at three different times after the accident in scenario version 1.

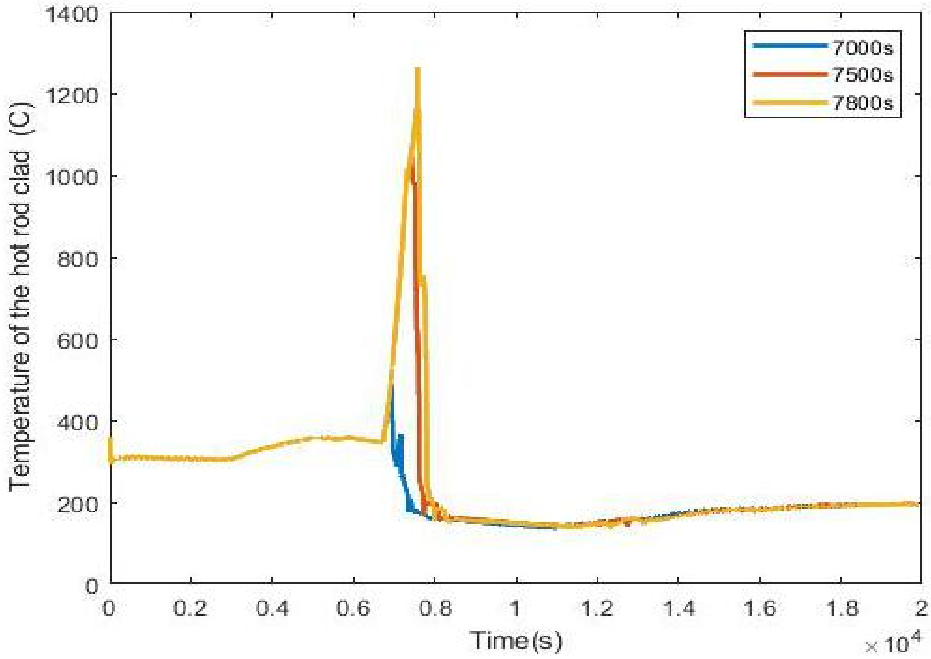


Fig 35. Temperature of the hot rod clad in the reactor emergency power recovery (2 train) at three different times after the accident in scenario version 2.

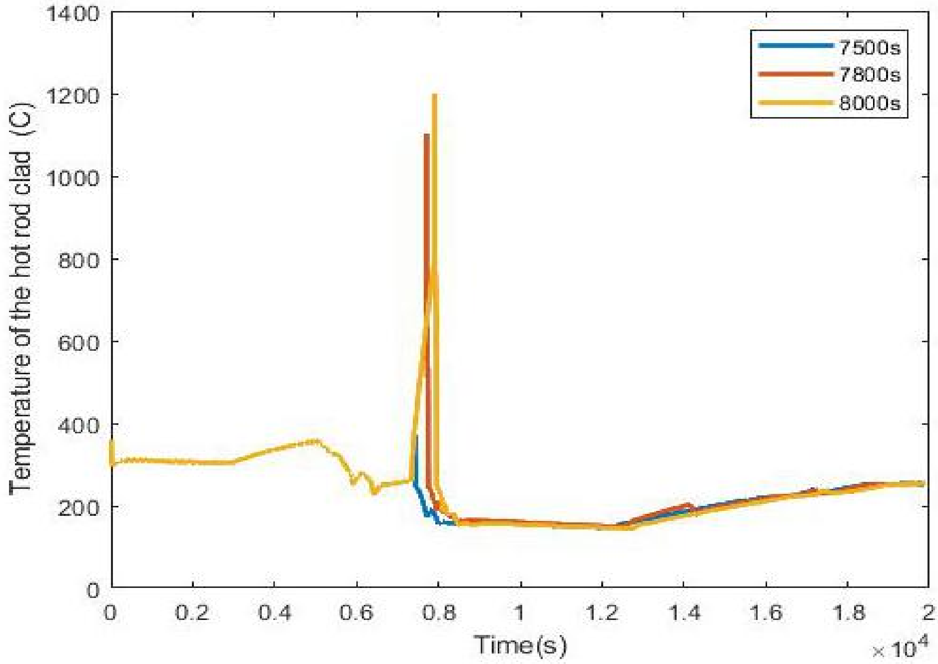


Fig 36. Temperature of the hot rod clad in the reactor emergency power recovery (2 train) at three different times after the accident in scenario version 3.

The steam generator's water level will keep rising until it reaches 2.2 meters due to the fluid injection shown in Figures 37–39. This scenario considers that two train safety systems are recovered, and two trains emergency feed water systems are working.

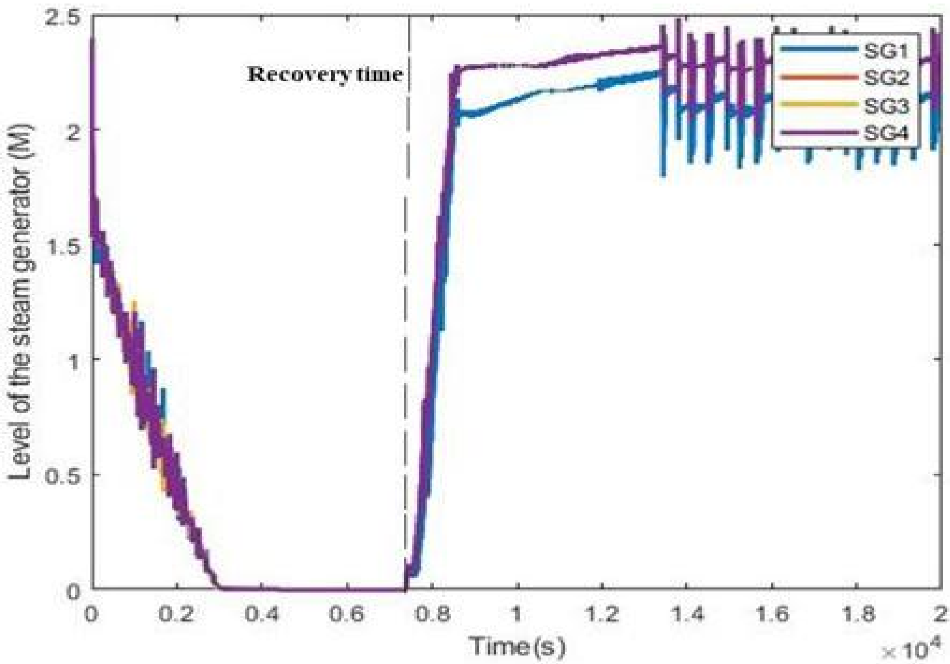


Fig 37. Steam generators water level changes in after emergency power restoration (2 train) in scenario version 1.

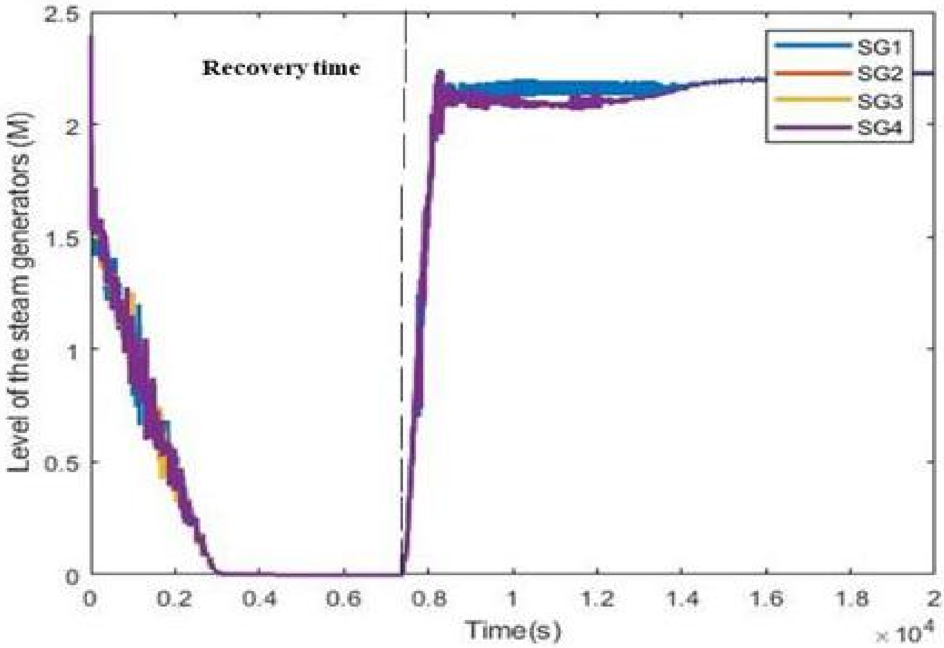


Fig 38. Steam generators water level changes after emergency power restoration (2 train) in scenario version 2.

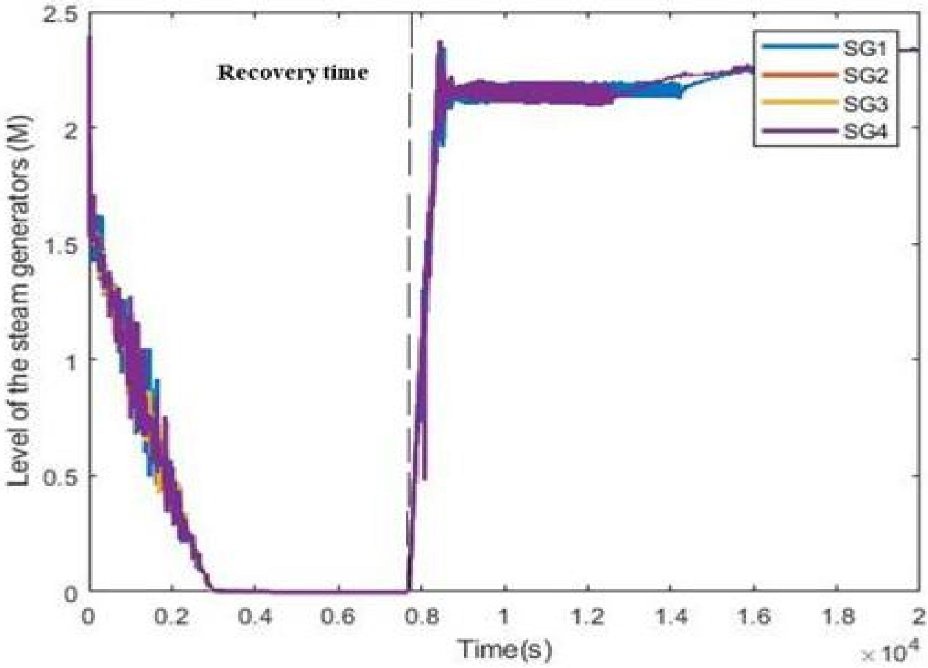


Fig 39. Steam generators water level changes after emergency power restoration (2 train) in scenario version 3.

The primary pressure decreases following the injection of EFW into SG, as depicted in Figures 40–42. In version 1, because the operator does not take any action, the activation of safety systems such as HPIS and fluid injection in the circuit will increase the pressure in the primary circuit. But considering that only two trains of safety systems are active and the amount of injection into the first circuit is limited, and also considering the activation of two EFW systems and heat exchange, the amount of pressure increase caused by injection into the primary circuit and the reduction of saturation temperature cooling are limited.

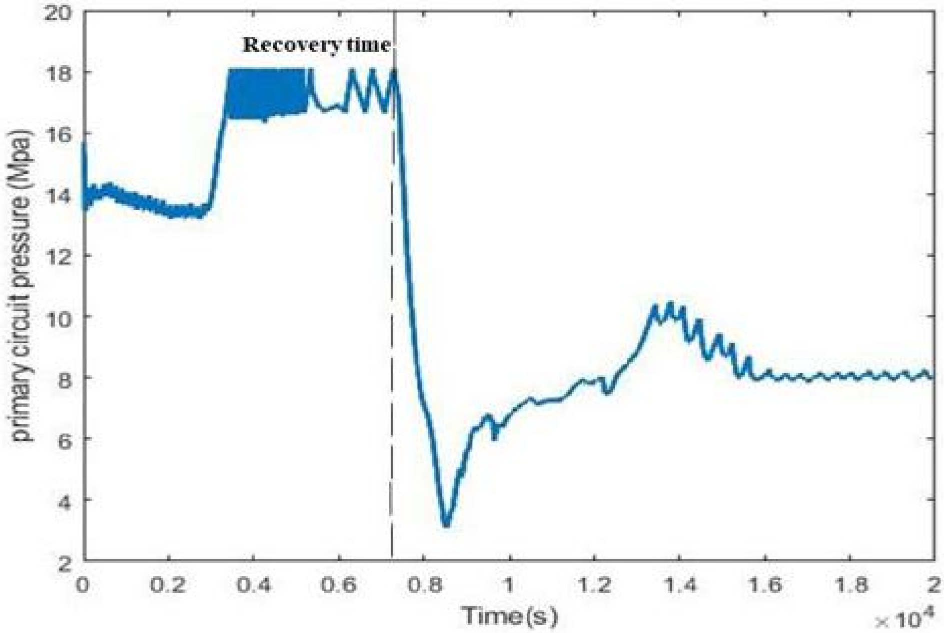


Fig 40. Primary circuit pressure after emergency power restoration (2 train) in scenario version1.

The primary pressure in version 2 decreases following the injection of EFW into SG, as depicted in Figure 41. This version considers that as the operator lowers the pressure in the primary loop with his actions, safety systems' injection in the primary loop, and effective heat transfer between the primary and secondary circuits, the primary pressure drops gradually.

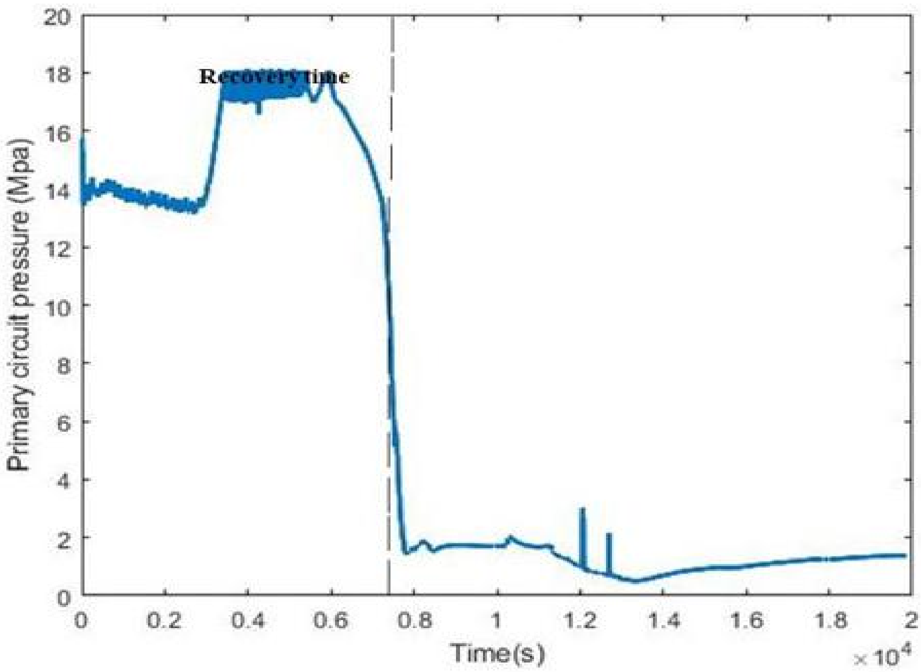


Fig 41. Primary circuit pressure after emergency power restoration (2 train) in scenario version2.

The operator has reduced the pressure of the primary circuit in version 3 to the activation threshold of the first and second-stage accumulators with various actions in this version of the scenario. The pressure will rise slightly due to the safety systems and the fluid injection into

the primary circuit, and then it will fall slowly again due to the heat transfer between the primary and secondary circuits, as shown in Figure 42.

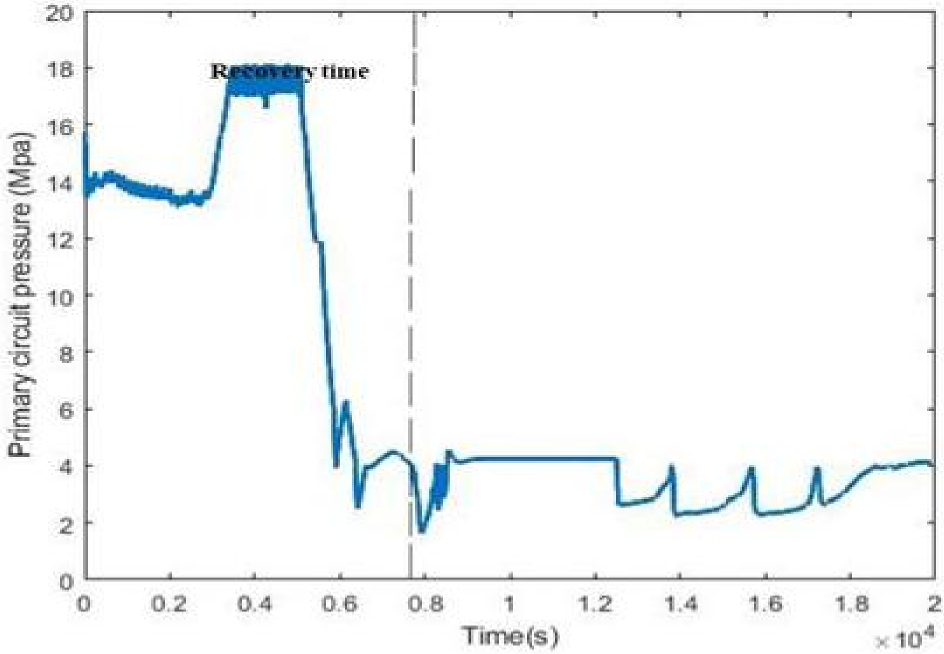


Fig 42. Primary circuit pressure after emergency power restoration (2 train) in scenario version3.

As the steam generator's water level rises in the end, the heat transfer between the first and second circuits increases, and the pressure in the second circuit will increase to the limit of the BRU-A valve's performance in version 1, shown in Figure 43.

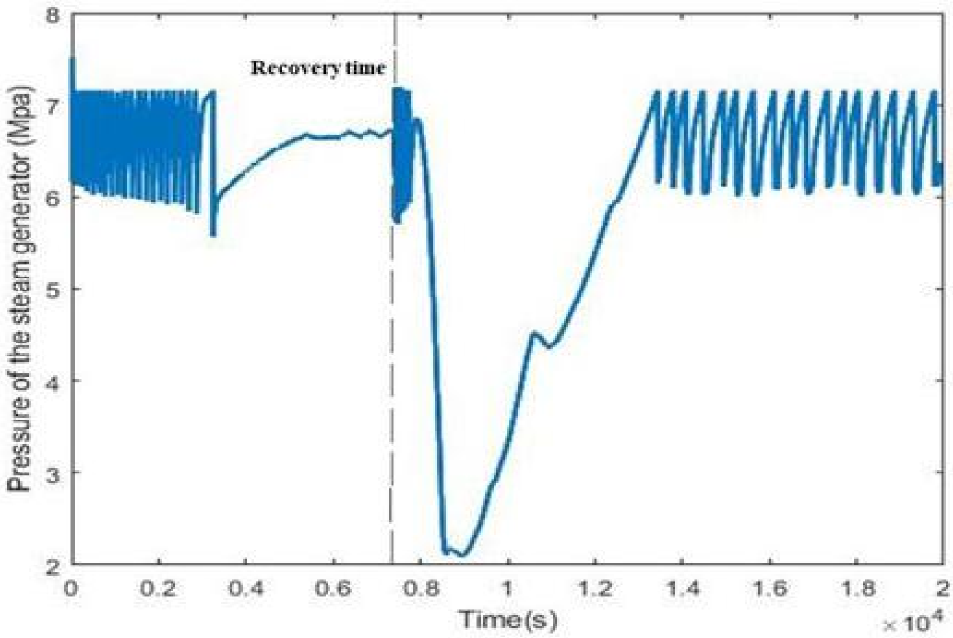


Fig 43. Steam generators pressure after emergency power restoration (2 train) in scenario version1.

As the steam generator's water level rises and the heat transfer between the first and second circuits increases, the pressure in the secondary and primary circuits will both decrease steadily due to the operator’s actions to lower the pressure (scenarios versions 2 and 3) in the

primary circuit's and the continuous downward trend in the pressure as shown in Figures 44–45.

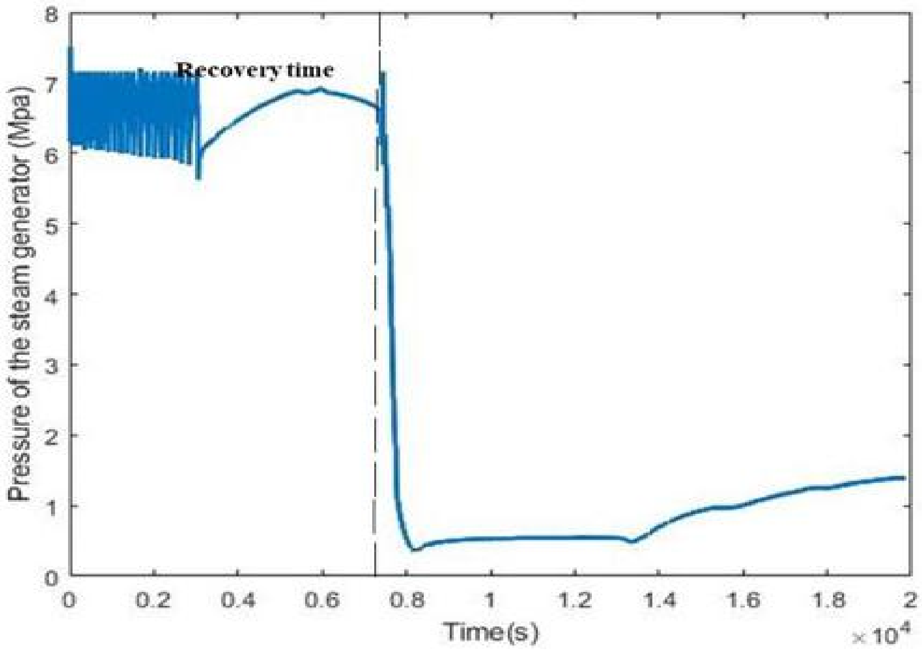


Fig 44. Steam generators pressure after emergency power restoration (2 train) in scenario version2.

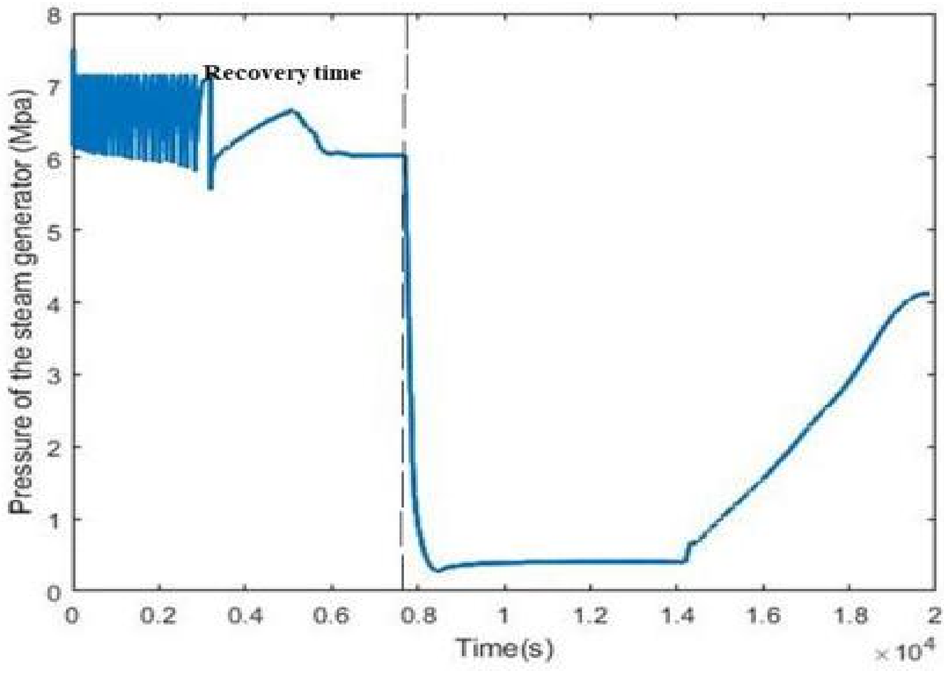


Fig 45. Steam generators pressure after emergency power restoration (2 train) in scenario version3.

6. Conclusion

The Fukushima accident highlighted the importance of evaluating a complete deterministic safety assessment of nuclear power plants to ensure they can withstand accidents and design extension conditions. In this research, which investigated the SBO accident in BNPP, which causes the highest core damage frequency, a validated Relap5 model of the BNPP was used for the analysis. In the first step, simulate the station blackout in different scenarios of the BNPP with and in the absence of operator actions. To delay the start of heating, the main

strategy of the operator is to reduce the initial side pressure and subsequently reach the activation point of the ACCs and KWU accumulators (version 3). However, the primary side pressure was lower than in the scenario version 1, which could lead to low-pressure core damage. Based on the deterministic safety analysis, the recovery time of diesel generators as well as the number of activated diesel generators and subsequently activated safety systems play an important role in accident control and management. According to the presented results, two scenarios have been considered. In the first scenario, it is assumed that diesel generators and, subsequently, all safety systems are activated. In this scenario, if the diesel generators are recovered before 7800 seconds and the operator reduces the primary side pressure by taking the necessary measures (scenario no. 3), it is possible to maintain and control the accident and prevent core damage.

In the second scenario, two trains of diesel generators (or portable generators) have been assumed, followed by two trains of safety systems. In this scenario, if the diesel generators are recovered before 7800 seconds and the operator decreases the primary side pressure by taking the necessary measures (scenario no. 3), it is possible to maintain and control the accident and prevent core damage. The proposed scenarios were reevaluated to ensure no fuel rod damage and adequate core cooling. Finally, considering the most conservative assumption, if it is only possible to recover two trains of the systems, the operator can prevent core damage by taking the necessary measures and decreasing the primary circuit pressure until the set time to recover the systems. Also, the management guidelines for the SBO accident are expanded for BNPP based on safety analysis. According to the results, to manage and control the accident, the operator must take the necessary measures to reduce the pressure in the first circuit. This can be done by using PSD valves (as outlined in scenario no. 3), and in this case, the operation of at least two trains of safety systems can prevent damage to the reactor core.

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